PNPS 1.1

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 8, 1971

Honorable Hastings Keith
House of Representatives

Dear Mr. Keith:

Attached for your information is a copy of an announcement concerning a change of date and location for prehearing conference on Pilgrim Nuclear Power Station at Plymouth, Massachusetts.

It is planned to mail this announcement to the news media today, ~~September 8~~. Oct 13th

Sincerely,

Robert D. O'Neill

Robert D. O'Neill, Director
Congressional Relations

Attachment:
As stated

PNPS 1.2

No. 0-161
Contact: Clare Miles
Tel. 973-3446
973-5371 (Copies)

Chuck Ives

FOR IMMEDIATE RELEASE

CHANGE OF DATE AND LOCATION FOR PREHEARING CONFERENCE ON PILGRIM NUCLEAR POWER STATION AT PLYMOUTH, MASSACHUSETTS

A new date and location for the first prehearing conference in the licensing proceedings on the Pilgrim Nuclear Power Plant at Plymouth, Massachusetts, has been set by the Atomic Safety and Licensing Board. The prehearing conference will be held at 10 a.m. on Wednesday, October 13, 1971, in the Memorial Hall, 83 Court Street, Plymouth.

The meeting, which is open to the public, had previously been scheduled for October 6 in the Probate Courtroom in the Registry Building in Plymouth.

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9/8/71

I welcome the hearings by the AEC on the licensing of Boston Edison~~'s~~ Co.'s Nuclear Power Plant in Plymouth. The hearings should provide a forum for an open discussion on the important issues of safety, and environmental protection.

Scientists and technical experts from the AEC, from Conservation groups, and from industry will be able to present in public their views to the hearing board, where they will be thoroughly examined and studied before a license is issued.

I am certain that all citizens want to have both a clean and healthy environment and an adequate economical electric power supply. For this reason, I hope that the hearing will proceed in an expeditious and business-like manner. If the plant is safe, there should be no delay in putting it into operation since the alternative could well be brown-outs or black-outs as well as more expensive electricity.

I am convinced that both environmental desires and realiable electric power supply can and must be achieved. I therefore urge all the parties to the proceedings to resolve the issues as speedily as possible.

Gus Wagner

PNPS 3.1

August 25, 1971

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

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APPENDICES

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1.0 INTRODUCTION

This report is the Atomic Energy Commission's safety evaluation of the application by the Boston Edison Company (BECO or applicant) for a license to operate the Pilgrim Nuclear Power Station (herein referred to as the facility). The facility is located on the western shore of Cape Cod Bay in the town of Plymouth, Plymouth County, Massachusetts; and is approximately 36 miles southeast of Boston, Massachusetts, and 44 miles east of Providence, Rhode Island.

In an Order dated August 26, 1968, the Atomic Safety and Licensing Board directed the Director of Regulation to issue to BECO a provisional construction permit for the Pilgrim Nuclear Power Station. The Commission thereupon issued Construction Permit No. CPPR-49 on that same date. The facility has been under construction since the date of this construction permit. [An exemption granted by the Commission on March 15, 1968, authorized certain below grade construction to proceed prior to issuance of the construction permit.]

The applicant submitted its Final Safety Analysis Report (FSAR), dated December 31, 1969, in support of its application for a facility operating license at a thermal power of 1998 MW which is about 4.5 percent increase in thermal power over that requested previously.

This safety evaluation report summarizes the results of the technical evaluation of the Pilgrim Station performed by the Commission's regulatory staff. Our evaluation included a technical review of the information submitted by the applicant with regard to the following principal matters:

1. We reviewed the population density and use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology and hydrology to determine that these characteristics had been determined adequately and had been given appropriate consideration in the plant design, and that the site characteristics were in accordance with the Commission's siting criteria (10 CFR Part 100) taking into consideration the design of the facility including the engineered safety features provided.

2. We reviewed the design, fabrication, construction, testing, and expected performance of the plant structures, systems, and components important to safety to determine that they are in accord with the Commission's General Design Criteria, other appropriate codes and standards, and the Commission's Quality Assurance Criteria, and that any departures from these criteria have been identified and justified.

3. We evaluated the response of the facility to various anticipated operating transients and to a broad spectrum of postulated accidents, and determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents to determine that the calculated potential off-site doses that might result in the very unlikely event of their occurrence would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.

4. We evaluated the applicant's plans for the conduct of plant operations, the organizational structure, the technical qualifications of operating and technical support personnel, the measures taken for industrial security, and the planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant is technically qualified to operate the plant and has established effective organizations and plans for continuing safe operation of the facility.

5. We evaluated the design of the systems provided for control of the radiological effluents from the plant to determine that these systems can control the release of radioactive wastes from the

station within the limits of the Commission's regulations (10 CFR 20) and that the applicant will operate the facility in such a manner as to reduce radioactive releases to levels that are as low as practicable.

During our review of the information submitted in the Safety Analysis Report we requested the applicant to provide additional information we needed for our evaluation. This additional information was provided in subsequent amendments to the application. In the course of our review we held numerous meetings with the applicant to discuss and clarify the technical information submitted. As a result of our review we requested a number of changes to be made in the facility design; these changes are described in the applicant's amendments and are discussed in appropriate sections of this report.

Many features of the design of this plant are similar to those we have evaluated and approved previously for other reactors now under construction or in operation. To the extent feasible and appropriate, we have made use of our previous evaluations to expedite our review of those features that were shown to be substantially the same as those previously considered. Where this has been done, the appropriate sections of this report identify the other facilities involved. Our safety evaluations of those other facilities have been published and are available for public inspection at the Atomic Energy Commission's Public Document Room.

The information submitted in the FSAR was supplemented by Amendments 13 through 30. All of these documents, except Amendment 25, are available for public inspection at the U.S. Atomic Energy Commission, Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Plymouth Public Library, North Street, Plymouth, Massachusetts. Amendment 25 contains solely the "Industrial Security Plan" and we have determined that it may be withheld from public disclosure under the Commission's Rules and Regulations 10 CFR 2.790(d) and 9.5(a)(4).

Our evaluation of the overall facility performance is based on the intent to operate the facilities at a thermal power level of 1998 MW, which will be the licensed power level. Although the construction permits for the facility indicated that the unit would be operated at an initial power level of 1912 MW(t), the safety evaluation for the construction permit review was also based on a power level of 1998 MW(t).

In addition to our review, the Advisory Committee on Reactor Safeguards (ACRS) reviewed the application and met with both the applicant and us to discuss the facility. The ACRS Report on the Pilgrim facility, dated April 7, 1971, is attached as Appendix A to this Safety Evaluation.

Based on our evaluation of the application to operate the facility, as presented in subsequent sections of this report, we have concluded that the Pilgrim Station can be operated as proposed at power levels up to 1998 MW(t) without endangering the health and safety of the public.

The construction permit was granted for a nuclear reactor rated at 1912 MWt; however, all safety systems were analyzed previously for the 1998 MWt power. The applicant now requests an operating license at thermal power levels up to 1998 MWt, a 4.5 per cent increase in thermal power. With this increase in design power level the thermal characteristics of the Pilgrim reactor core are substantially the same as those of the Dresden 2 and 3, Monticello, and Millstone 1 reactors that have been licensed previously.

2.0 SITE

2.1 Site Description

The Pilgrim Nuclear Power Station is located within the corporate limits of the Town of Plymouth, Plymouth County, Massachusetts, and approximately 4.4 miles east of the Plymouth Rock Monument. It is on the western shore of Cape Cod Bay and just south of Plymouth Bay at a place called Rocky Point.

The 517 acre site can be described roughly as a rectangle about 0.45 miles wide by 1.8 miles long, with the long axis oriented northwest to southeast parallel to the Cape Cod Bay shore. The Pilgrim Station is located on the northwestern end of the site abutting the shoreline. The centerline of the reactor is approximately 600 yards from the nearest land property boundaries in three directions; i.e. WNW, SSW, and SSE. The offgas stack is located northwest of the reactor building and about 350 yards inside the nearest property line.

The applicant has defined the exclusion area as the northwestern end of the Boston Edison Company property, not including the small triangular plot, about 12 acres in area, positioned approximately in the center of the site, that is the property of a nearby resident. Rocky Hill Road traverses the exclusion area, and recreational activities may be allowed in well defined areas within the exclusion area. Public access into these areas does not interfere with normal operations of the facility, and the applicant has provided appropriate and effective arrangements to control traffic on this road and access to these areas in an emergency.

The dominant topographic feature in the locality is the Pine Hills, a ridge west of the site, which runs from its northern boundary at Plymouth Bay to the south for about 4 miles. From the shoreline on the northeast side, the land rises to a maximum onsite height of about 280 feet above Mean Sea Level (MSL) at Cleft Rock. The maximum offsite height of about 400 feet occurs on Manomet Hill of the Pine Hills about 1.5 miles southwest of the Pilgrim Station.

More than half the total area within 50 miles of the site is ocean water, which is used for commercial seafood harvesting and recreational activities. The land area within 10 miles of the site is characterized as 85% open space, 7% agricultural (mainly cranberry bogs), and 8% residential and commercial.

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The applicant has described his "low population zone" as defined in the Commission's Rules and Regulations, 10 CFR 100.3, as that area within a radius of 4.25 miles from the reactor. The total 1965 resident population within this low population zone is estimated at about 4,000 residents, and by the year 2015 it is predicted to be about 7,700 by the applicant. Because of the proximity of the site to coastal summer resort areas, the population varies significantly with the seasons.

The nearest community, Priscilla Beach, is southeast of Pilgrim Station, and several other beach communities are located on the shoreline to the southeast. Except for these beach communities and residential area southeast of the Plymouth business district, the land area within the low population zone is sparsely developed. It is expected that the population within this region will approximately double over the life of the plant.

The "population center distance" is given by the applicant as approximately 23 miles, the distance from the Pilgrim Station to the nearest boundary of Brockton, Massachusetts, a densely populated center containing more than 25,000 residents.

Based on our evaluation of the population data, and of the calculated potential doses that might result from the postulated design basis accidents (presented in Section 9.0), we conclude that the distances established for the exclusion zone, the low population zone boundary

and the nearest population center meet the guidelines given in 10 CFR Part 100.

2.2 Meteorology

The semipeninsular location on the coast of Massachusetts is the principal determinant of the major meteorological characteristics of this site. Historical records from Plymouth indicate that northeastern temperature extremes are moderated by the nearby ocean heat sink. Records at Plymouth indicate total precipitation is fairly uniform from month to month and neither annual rainfall nor snowfall is unusually high.

The storm cycle in the area consists of thunderstorms in spring and summer, possible hurricanes in late summer and fall, and northeasters in the winter. Severe tornado activity is not common in eastern Massachusetts, and the proximity to the ocean and the terrain in the vicinity of the site are unfavorable to such activity.

The prevailing westerlies generally result in relatively good diffusion characteristics for gaseous effluents routinely released from the station. Under these conditions gases discharged from the 400-ft-high off-gas stack are dispersed by their elevated release and mixing as they travel out over Cape Cod Bay. The applicant has been conducting an onsite meteorological program since May 1968. The program involves

the use of a 220-ft-high tower. Data from this program include measurements of wind speed, wind direction and temperature at various tower elevations. We and our consultant, the Environmental Science Services Administration, of the Air Resource Environmental Laboratory have reviewed the applicant's program and results. The consultant's report is attached as Appendix B.

In our review of the meteorology for this station, we have investigated the occurrence of fumigation conditions at this site. In the spring and summer, when the water surface is relatively cooler than the land mass, the combination of stable lower air, onshore winds and differential heating as the seabreeze moves inland can produce fumigation conditions. Under fumigation conditions the stack effluents may encounter a violent vertical mixing with surface winds at some point downwind of the stack, resulting in a higher concentration of stack effluents at ground levels than would otherwise occur. The occurrence of seabreeze-fumigation conditions has been investigated by a special smoke release study conducted by the applicant during the summer of 1969. Although seabreeze-fumigation conditions occur infrequently and are of short duration, in our accident analyses we have assumed conservatively that fumigation conditions exist for the initial two-hour period subsequent to postulated accidents.

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We conclude that the meteorological characteristics of the site have been determined adequately and provide an acceptable basis for determining routine gaseous effluent release limits, and for establishing a conservative meteorological model for use in the accident evaluations described in Section 9.0 of this evaluation.

2.3 Hydrology

The topography of the site is such that surface water drainage will be toward Cape Cod Bay. The ground-water-table follows the topography with flow toward the Bay. The nearest public well is about 2-3/4 miles southeast of the station, and there is a private well about 800 yards away at the southeast property boundary. Even in the unlikely event of an accidental spill of radioactive liquids, there is little or no potential for contamination of wells in the neighboring areas because of the favorable drainage characteristics associated with the station site.

Cape Cod Bay is the principal oceanographic feature of the site and acts as the source and receptor of station cooling water. It is approximately circular in shape with a diameter of about 20 nautical miles. It is bounded from the northeast to the southwest by Cape Cod and on the west by the Massachusetts mainland, with a mouth across the northern boundary about 17-1/2 nautical miles across. Cape Cod Bay has a surface area of approximately 430 square nautical miles, and

attains a depth of about 180 feet at its mouth. The southward coastal current enters the Bay along the western shore and leaves the Bay on the eastern end of its mouth, resulting in a counter clockwise circulation. The flushing rate of the Bay is dependent on the general circulation pattern and on the influences of tides and winds.

Severe coastal storms, both hurricanes and northeasters, can be expected in the area. Breakwaters have been built to protect the intake channel and exposed station structures. The top of the breakwater is at an elevation of 11.2 feet above mean sea level (MSL). The applicant calculated a probable maximum surge height (still water level) of 13.5 feet MSL. We and our consultant, Coastal Engineering Research Center, made an independent analysis by use of probable maximum hurricane parameters applied to the bathystrophic storm tide theory. This analysis predicted a probable maximum surge height of 14.8 feet MSL. We have also reviewed the wave action associated with this storm, including the wave runup effects, and have concluded that the station elevation (20.0 feet MSL with minimum entrance elevations for all structures at 23.0 feet MSL) is adequate to prevent station damage or shutdown at the higher probable maximum surge height.

Storm-driven low tides can reach an elevation of -10.1 feet MSL. Although this would result in station shutdown through loss of the main circulating water pumps, the salt water service water pumps would remain

operative and provide water for the shutdown cooling requirements of the station.

In conclusion, neither storm flooding nor low tide conditions would result in a situation that might prevent safe shutdown of the Pilgrim Station.

2.4 Geology and Seismology

We and our consultant, the U.S. Geological Survey, evaluated the geological aspects of the site during our review of this facility, prior to issuance of a construction permit. Additional study of the geological and seismological characteristics of the site since the construction permit was issued has confirmed the original conclusions regarding the site.

There are no known geological faults at or near the site and none were revealed by the drilling, excavation, or geophysical site investigations. The nearest mapped fault is located about 17 miles west of the site in the Narragansett Basin. Although there are no identified faults or other geologic structures that might localize earthquakes in the immediate vicinity of the site, details regarding either local or regional geology are not well known, due to the low relief and the depth of glacial deposits that obscure bedrock throughout much of the region. Since the epicenters of earthquakes that have occurred in the region

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cannot be related to any known geologic structures, it is assumed that earthquakes of intensities characteristic of the southeastern region of Massachusetts south of the Boston Basin also may occur at the station.

A strong motion accelerograph and three peak-reading accelerographs will be installed to record accelerations of the ground and key structures in the event of an earthquake at the site. These data would be employed in the subsequent evaluation of the effects of the earthquake on the safe operation of the facility. The seismic aspects of the facility design are discussed in Section 10.0 of this evaluation.

2.5 Environmental Radiation Monitoring

In August 1969, the applicant initiated an environmental radiation monitoring program that will continue during plant operation. The requirements for the environmental radiation monitoring program are listed in the Technical Specifications. The program provides that samples be taken from air and principal aquatic and terrestrial items of the area at appropriate locations and frequencies. Included are samples to be taken from domestic water stations, sea water, marine life, bottom sediment, milk, appropriate food crops and cranberries.

Detailed records will be kept of radiological discharges from the plant and periodic audits of the operation will be made by the AEC's Division of Compliance. The program was developed with the cooperation

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of the U.S. Fish and Wildlife Service and the Massachusetts Department of Natural Resources. The report of the U.S. Fish and Wildlife Service is attached as Appendix C; and the Boston Edison Company's response to that report is attached as Appendix D.

We conclude that the applicant's program is adequate for monitoring the radiological aspects of plant operation on the environs and assessing the health and safety aspects of the release of radioactivity to the environment from the operation of the plant.

3.0 NUCLEAR STEAM SUPPLY SYSTEM

3.1 General

The nuclear steam supply system, initial nuclear fuel loading, and turbine-generator were designed and manufactured by the General Electric Company. The Bechtel Corporation provided engineering and construction services for the design and construction of the station.

The reactor is a single cycle, forced circulation, boiling water reactor designed for a thermal power of 1998 Mwt and a gross electrical output of approximately 687 MWe. The principal nuclear steam supply components, station design features, materials of construction and architectural arrangement of various systems of the Pilgrim Station are similar to those of the Millstone I facility which is currently in operation. In many features the Pilgrim Station is also similar to other operating BWR plants including Dresden Units 2 and 3 and Monticello. These plants

and Pilgrim are designed to similar reactor core power densities; they all use internal jet pumps to recirculate coolant through the reactor core; their reactor and primary containment vessels are designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code; they utilize the vapor suppression type of containment; and their emergency core cooling systems are similar in functions and capabilities. Because of the similarity in design features between Pilgrim and these other nuclear power stations, we have relied in part on our earlier reviews of the similar plants in reaching conclusions regarding the adequacy of such features in the Pilgrim Station.

3.2 Reactor Design

3.2.1 General

The reactor core contains 580 fuel assemblies, each of which consists of a 7 x 7 square array of cylindrical fuel rods enclosed within a Zircaloy-4 fuel channel. The fuel rods consist of low-enrichment, sintered UO_2 fuel pellets clad in Zircaloy-2 tubes. The outer diameter of each fuel rod is 0.563 inch; the fuel pellets have a diameter of 0.488 inch, and are stacked end-to-end within the tube to provide an active fuel height of 144 inches.

In the reactor core, the fuel assemblies are grouped in modules of four fuel assemblies. A cruciform blade control rod moves vertically in the interstice between the assemblies in each module. The reactor

core contains 145 moveable control rods of the bottom-entry, upward scram type, moved vertically within the core by hydraulically operated drives of the same design as used on other GE-BWR plants previously licensed.

The numbers of fuel assemblies, moveable control rods, and other components setting the reactor core size are identical with Millstone I. The fuel assemblies, control rods and other core components are dimensionally identical to those used in the Dresden 2 and 3, and Monticello reactor cores. Pilgrim differs from the other four cores in using four different fuel enrichments in each fuel assembly. The increase from three to four enrichments produces slightly lower rod-to-rod peaking factors within the fuel assembly for the initial fuel loading. We have concluded that the minor differences in the core characteristics and component designs have no significant safety implications and, on the basis of our earlier review of the Millstone I reactor core and the similarity of components and their design limitations as reviewed for the Dresden Units 2 and 3, and Monticello reactors, we conclude that the Pilgrim core design is acceptable.

3.2.2 Core Thermal and Hydraulic Design

The reactor core design power level during the construction permit review was 1912 MWt. The applicant now proposes to operate the Pilgrim reactor at thermal power levels up to 1998 MWt, a 4.5 per

cent increase in thermal power. With this increase in design power level, the thermal characteristics of the Pilgrim reactor core are substantially the same as those of the Dresden 2 and 3, Monticello, and Millstone I designs. In all cases the critical heat flux limits are based on data contained in the GE topical report, APED-5286, "Design Basis for Critical Heat Flux Conditions in Boiling Water Reactors."

Our review of the applicant's analyses of the various transients that can be expected to occur during the lifetime of the plant, indicated that the analyses are the same as those previously approved for Dresden Units 2 and 3 and other similar BWR plants. The core thermal and hydraulic design basis is to control the local power density within the core to levels that assure that the fuel heat flux is maintained within acceptable limits so that the fuel rods do not overheat during normal plant operation including operational transients.

The controlling mechanism that could cause fuel damage in reactor transients is severe overheating of the fuel cladding caused by inadequate cooling if critical heat flux conditions in the core are exceeded. The critical heat flux is defined as that which occurs on the fuel cladding at the onset of the transition from nucleate boiling to film boiling. For design purposes the critical heat flux is conservatively used as a fuel thermal limit although actual fuel damage may not occur until well into the film boiling regime. The present critical heat flux limits are

presented in the GE topical report APED-5286, "Design Basis for Critical Heat Flux Conditions in Boiling Water Reactors", issued in 1966. The minimum critical heat flux ratio (MCHFR) is defined as the ratio of the actual critical heat flux to the maximum calculated heat flux occurring at any period during operation including reactor anticipated transients. A MCHFR > 1.0 conservatively assures that cooling of the fuel is maintained through nucleate boiling heat transfer.

The current design basis for normal operation is that the MCHFR calculated for any point is greater than 1.9 during normal operation and greater than 1.0 during anticipated transients. These limits provide considerable margin between expected conditions and those required to cause fuel clad damage since the critical heat flux correlation presented in APED-5286 is conservatively based on a limit line drawn below all of the available experimental data points.

We have reviewed the methods used to calculate the MCHFR, the experimental basis for the calculation, its validity as a damage limit and the applicant's analyses of normal operation and anticipated transients for this station and previously reviewed reactors, and conclude that the design provides adequate margin to protect the core against fuel damage.

3.2.3 Reactivity Control

Reactor power can be controlled by either movement of control rods or variation in reactor coolant recirculation system flow rate.

Temporary fixed control curtains are installed to supplement the moveable control rods to limit the reactivity of the core. A standby liquid control system is also provided as a backup shutdown system.

Control rods are used to bring the reactor through the full range of power (from shutdown to full power operation), to shape the reactor power distribution, and to compensate for changes in reactivity resulting from fuel burnup. Each control rod drive has separate control and rapid insertion (scram) devices. A common hydraulic pressure source for normal operation and a common dump volume for scram operation are used for the drives. On the basis of our review of the drive system design and the supporting evidence accumulated from operation of similar systems in other General Electric reactors, we conclude that the installed system will meet the functional performance requirements for Pilgrim in a safe manner.

During operation at power levels below 10% of the rated power, control rod reactivity worths are limited by the rod worth minimizer (RWM), a device which utilizes a computer to restrict control rod patterns such that the total worth of the rods that can be moved will be no more than 1% Δk. For reactor power levels in excess of 10% of the rated power, the maximum worth of any control rod that could be established is 3.8% Δk. Calculations of the consequences of a control-rod-drop accident (where a control rod equipped with a velocity limiter is assumed to fall by

gravity from the core region with a rod worth of 3.8% Δk and the reactor power is in excess of 10% of the rated power) indicate that the peak fuel enthalpy is about 200 cal/gm, which is less than the enthalpy required for incipient fuel melting. Accordingly, we have concluded that use of the RWM is not required at power levels above 10%. Nevertheless, the Technical Specifications require that control rod patterns be established so that the maximum worth of any operable control rod shall be less than 2.5% Δk.

A control-rod-ejection accident is precluded by a control rod housing support structure located below the reactor pressure vessel, similar to that installed on the other large General Electric reactors. This structure limits the distance that a ruptured control rod drive housing could be displaced. The applicant concluded and we agree, that the control rod displacement would be so small in this event that any resulting nuclear transient could not be sufficient to cause fuel rod failure.

Reactor power can also be controlled through changes in the primary coolant recirculation flow rate. The recirculation flow control system is the normal control method used to adjust reactor power level to station load demand whenever the reactor is operating between approximately 60% to 100% rated power. The recirculation flow control system is designed to allow either manual or automatic control of reactor

power. This method of reactor power control has been demonstrated in the Dresden Units 2 and 3, Monticello and Millstone I facilities.

The standby liquid control system is designed to bring the reactor to a cold shutdown condition from the full power steady-state operating condition at any time in core life, independent of the control rod system capabilities. The injection rate of the system is adequate to compensate for the effects of xenon burnup.

Each of the foregoing design features is similar to the corresponding features provided in plants we have previously reviewed. On the basis of our previous review and of satisfactory operating experience, we conclude that the mechanical, thermal and hydraulic, and reactivity control features of the Pilgrim reactor are acceptable.

3.3 Reactor Coolant System

3.3.1 General

The principal components of the reactor coolant system include the reactor vessel, the reactor vessel internals, the two recirculation pumps and lines, the main steam and feedwater lines, the pressure relief system, and portions of the primary coolant auxiliary systems, i.e., the reactor core isolation system (RCIC), the residual heat removal system (RHR), and the reactor water cleanup system. Portions of these systems as well as other piping extend from the reactor vessel up to

the second isolation valve. All components of the system were designed to applicable codes in effect at the time the components were ordered.

3.3.2 Reactor Pressure Vessel

The reactor pressure vessel was designed, fabricated, inspected and stamped in accordance with the requirements for Class A vessels of the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and January 1966 Addenda. The vessel was fabricated by Combustion Engineering, Incorporated, Chattanooga Division.

The reactor vessel originally contained sensitized stainless steel components. Stainless steel that has been sensitized has an increased susceptability to stress-corrosion cracking. In Amendment No. 16, the applicant indicated that the Pilgrim reactor vessel would not use furnace sensitized stainless steel directly exposed to the reactor coolant environment. Thus all stainless steel safe ends originally installed on the reactor vessel that might have become sensitized during heat treatment of the vessel were removed and replaced with new unsensitized stainless steel safe ends. Furnace sensitized attachment pads on the inside vessel surface for the jet pump riser brace and recirculation inlet thermal sleeve attachment were weld overlaid to isolate these load bearing members from the reactor coolant. These actions reduce the probability of stress corrosion cracking of sensitized stainless steel components. We have reviewed the nozzles and weld pads affected, and

the welding procedures used by the applicant for this work. The repair program performed on the vessel is adequate to provide the necessary assurance that the reactor vessel would be protected against the effects of corrodents that could lead to stress corrosion cracking that has been observed on other pressure vessels where sensitized stainless steel components were used. We conclude that the steps taken to eliminate furnace sensitized stainless steel from the reactor vessel are acceptable.

We have reviewed the fracture toughness data for the ferritic materials within the reactor vessel considering that recent test data on this subject indicate that current ASME Code rules (used by the applicant) may not always be sufficiently conservative, and may not assure adequate fracture toughness of these materials. We have accordingly established conservative limits on pressure-temperature relationships for reactor startup and shutdown operations; i.e., the pressure within the reactor coolant pressure boundary shall not exceed 250 psig at temperatures below 180° F at any time fuel is in the reactor vessel. This limitation has been made a Technical Specification requirement.

The applicant's material surveillance program is consistent with programs that have been accepted on previous similar BWR plants, and we conclude that this program is adequate to monitor effects of radiation to the reactor vessel.

3.3.3 Reactor Vessel Internals

The reactor internals are designed as Class I (seismic) components to function within the stress limit criteria of Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, for all design loading conditions of mechanical, hydraulic and thermal origin including anticipated plant transients and the operational basis earthquake (OBE).

The reactor internals design provides for the maintenance of a coolable core configuration and for safe shutdown of the plant under the loads which would result from the combination of operating loads plus design basis earthquake (DBE) loads, or operating loads plus DBE plus the design basis accident (DBA) loads. The primary stress limits applied to the design of the reactor internals under the above loading combinations comply with the acceptable emergency and faulted stress limits specified in the component codes. The design assures that deflections of the fuel channels, control rod housings and core support structure under the above loading combinations will not prevent control rod operability and will assure the preservation of core cooling geometry.

We find that the design loading conditions, calculated design stresses, deflection limits, and design fatigue analyses as applied to the Pilgrim reactor vessel internals are acceptable and that adequate margins of safety are available to provide reasonable assurance that core cooling capabilities will not be impaired under the most severe loading conditions.

The applicant will perform a vibration test program on the reactor internals during preoperational and startup testing. The reactor internals for Pilgrim are substantially identical to components and structure that have been installed in earlier BWR plants.

The data obtained from the vibration test program will be adequate to determine the vibrational characteristics of the jet pump risers and other reactor vessel internals to confirm that there are no excessive vibratory motions. We conclude that the applicant's program is acceptable.

3.3.4 Reactor Coolant System Piping

Piping within the reactor coolant system was designed, fabricated, and inspected to meet the requirements of the USA Standard Code for Pressure Piping, Power Piping, USAS B31.1.0 - 1967, and supplemental requirements as given in Appendix A; Pressure Integrity of Piping and Equipment Pressure Parts of the FSAR. The reactor coolant system within the primary containment was designed as a Class I (seismic) system to withstand normal design loads of mechanical, hydraulic and thermal origin, including anticipated transients and the operational basis earthquake (OBE) and design basis earthquake (DBE). We conclude that the piping design is acceptable.

The applicant has chosen two methods to provide protection of the primary containment against possible damage due to pipe whip in the unlikely event of a large pipe rupture. The recirculation loops are provided with a system of supports designed to limit pipe motion so that reaction forces associated with any split or circumferential break do not jeopardize containment integrity. Similarly, large pipes which penetrate the containment are designed with anchors or limit stops located outside the containment to limit the movement of the pipe. These stops are designed to withstand the jet forces associated with the clean break of the pipe and thus maintain the integrity of the containment. In the case of the main steam piping, feedwater piping, residual heat removal system piping, and high pressure coolant injection system steam line piping within the containment, the applicant has provided protection against pipe whip by installing protective structures on the interior face of the drywell. These structures are located to protect the containment integrity where a large pipe weld failure could result in pipe movement to the extent that the ruptured pipe could contact the interior of the drywell shell with sufficient energy to perforate the drywell. These protective structures are provided in the spherical portion of the drywell only, since pipe ruptures within the cylindrical section of the drywell do not result in the impact energies required to perforate the drywell shell. We have reviewed the applicant's design criteria for protecting the primary

containment integrity against pipe whip and conclude that they are acceptable.

The main steam lines inside the primary containment are equipped with flow restrictors located close to the reactor vessel. The purpose of the restrictors is to limit the quantity of steam that may be discharged in the event of steam line rupture. The restrictor is a simple venturi welded into each steam line between the reactor vessel and first isolation valve. The restrictors have no moving parts and are similar to those provided on other BWR facilities.

The capability of the biological shield surrounding the reactor vessel to withstand the internal pressure and jet impingement loads that could be developed in the event of a failure in the region of a nozzle on the reactor vessel was analyzed by the applicant and the shield wall was shown to be capable of withstanding a differential pressure of 54 psi. The pressures and jet loads that might be developed by such a rupture within the biological shield were not sufficient to exceed an equivalent differential pressure of 54 psi. The applicant states that all shield plugs in the biological shield (provided to permit inservice inspection of these nozzle safe-ends) will be restrained so that the annulus pressure of 54 psi, due to combined static pressure and appropriate jet impingement pressures, would not cause the shield plugs to become missiles. We conclude that the combination of static pressure

and jet impingement loads resultant from a reactor vessel safe-end nozzle failure cannot lead to failure of the biological shield with unacceptable consequences.

3.3.5 Pressure Relief System

Overpressure protection for the Pilgrim reactor vessel is provided by safety valves and relief valves similar to those used at Millstone I and other BWR plants. The safety valves, two in number, are a balanced, spring-loaded type. The four relief valves for the Pilgrim reactor vessel are dual-purpose valves consisting of two main sections: the pilot valve section and the main valve section. The valves are self-actuating at the set-relieving pressure and also are pilot-operated which permits remote manual actuation or automatic relief at pressures below the set-point. For this latter function, each valve contains an air-powered diaphragm actuator capable of opening the valve and holding it open. This automatic depressurization feature is associated with the provisions for emergency core cooling. We conclude that the safety and relief valve systems, when supplemented by the action of the reactor protection system, provide adequate protection against over-pressurization of the reactor coolant boundary.

3.3.6 Primary Coolant Auxiliary Systems

The primary coolant auxiliary systems consist of the reactor core isolation cooling system (RCIC), the reactor shutdown and torus water

cooling modes of the residual heat removal system (RHR), the reactor water cleanup system, and the main steam line and feed-water piping. The primary coolant auxiliary systems are designed as Class I (seismic) systems except for the reactor water cleanup system external to the primary containment and the main steam lines and feedwater lines outside the primary containment. All piping in these systems is designed and fabricated to the requirements of the Power Piping Code, USAS B31.1.0 - 1967. We have reviewed the design, fabrication and inspection requirements used for those piping systems and find them acceptable. We have also reviewed the various codes and specifications used for the design, fabrication and inspection of the various tanks and heat exchangers included in these systems and found them acceptable for their respective applications.

The Reactor Core Isolation Cooling System (RCIC) is of the same design and serves the same function as in the Monticello, Quad-Cities and Vermont Yankee designs. The function is to supply about 400 gpm of water to the reactor vessel so that the core does not become uncovered in the event that the vessel is isolated from the feedwater system. This condition would occur in the event of a loss of all offsite power. In this case, upon isolation of the reactor, the relief valves and the RCIC system would be manually actuated so as to remove the core decay heat through blowdown of steam and concurrently to maintain water

level. If not manually actuated, the RCIC system will be automatically actuated when blowdown through the relief valves causes the water level to reach a low water level set point.

The Residual Heat Removal System (RHR) consists of two interconnected low pressure cooling loops connected to the primary coolant recirculation loops by a single suction line and return lines to the reactor inlet side of each recirculation loop. Each loop contains two pumps in parallel and a single heat exchanger. In addition to its function as a Core Standby Cooling System, the RHR provides a means of removing residual heat from the nuclear system so that refueling and servicing can be performed. The RHR may also provide cooling to the suppression pool for RCIC operation, and may be used to supplement the fuel pool cooling system when necessary.

The Reactor Water Cleanup System provides a means to maintain high reactor water purity to limit chemical and corrosive action within the primary coolant system, to remove corrosion products in the reactor coolant and thereby limit impurities available to neutron flux activation and for decreasing reactor water inventory during heatup.

The main steam line and feedwater piping systems provide for the routing of the reactor steam to the main turbine generator and the supply of reactor makeup water respectively. The main steam lines are fitted

with flow-restrictors to reduce the rate of coolant loss in the event of a main steam line rupture outside of the primary containment. Main steam line isolation valves both inside and outside the containment provide a redundant means of quickly terminating steam blowdown during this accident.

We have reviewed these features and systems on the basis of their similarity to those we reviewed for other reactors now in operation and conclude that they are acceptable for the Pilgrim facility.

3.3.7 Leak Detection

Leakage from the reactor pressure boundary inside the primary containment is measured by monitoring floor and equipment drain sumps and by measurement of temperature, pressure and humidity of the containment atmosphere. These methods are capable of detecting leakage in excess of approximately 5 gpm. We concluded that a more sensitive technique should be provided, and the applicant now proposes to monitor the containment atmosphere for airborne radioactivity. This system will be capable of monitoring three areas inside the primary containment for particulate gaseous and halogen radioactivity. The areas to be monitored include the annular space between the reactor vessel and biological shield, and each recirculation pump area. Based on operating experience at other nuclear power plants and an analysis by the applicant, a sensitivity of less than 1 gpm is anticipated. We conclude that the Pilgrim facility has an adequate primary coolant leak detection system.

3.3.8 Inservice Inspection Program

The applicant is providing access to the primary coolant boundary in compliance with Section XI of the ASME Code. The applicant will reevaluate the program after 5 years of reactor operation and report the results. In addition, modifications to the program would be proposed for our review.

The required inservice inspection program for the primary coolant system is included in the Technical Specifications. Boston Edison plans to inspect safety related systems beyond the primary coolant boundary through visual inspections. The access provisions for these visual inspections meet our current requirements for inservice inspection of safety related systems. We conclude that Pilgrim has an acceptable inservice inspection program.

4.0 CONTAINMENT SYSTEMS

4.1 Primary Containment

4.1.1 Description and Design

The primary containment is a typical "lightbulb" pressure suppression system consisting of a drywell, pressure suppression chamber (torus), and a connecting vent system. The drywell has a steel spherical lower portion 64 feet in diameter, and a steel cylindrical upper portion approximately 34 feet in diameter. Overall height of the drywell is about 110 feet. The pressure suppression chamber is a steel torus

below and encircling the drywell, with a centerline of approximately 102 feet and a cross-sectional diameter of 29.5 feet. Eight vent pipes lead from the drywell to a header inside the torus, and 96 downcomer pipes project downward from the header terminating approximately 4 feet below the surface of the torus pool. The free air volumes in the drywell and torus are approximately 147,000 ft^3 and 120,000 ft^3 respectively. The torus pool is required to contain a minimum of 84,000 ft^3 of water whenever the reactor vessel is pressurized. In the event of a reactor coolant system pipe rupture within the drywell, the released steam passes through the vent pipes, torus header, and downcomer pipes into the torus pool water where it is condensed. This transfer of energy into the pool water reduces the peak accident pressure that otherwise would be experienced by the primary containment.

The applicant has calculated that the peak pressures that might be reached as a result of the design basis loss-of-coolant accident are 45 psig in the drywell and 27 psig in the torus. These pressures were calculated assuming a hypothetical instantaneous break of one recirculation loop pipeline, all valves in the recirculation system open, and unobstructed flow from both ends of the severed pipe. The analytical methods used are the same as those used on other recently reviewed BWR plants and have been checked by comparison with the results of tests performed at the Moss Landing test facility.

The primary containment is designed for an internal pressure of 56 psig coincident with a temperature of 281°F. In accordance with the ASME Boiler and Pressure Vessel Code, Section III, maximum drywell pressures up to 62 psig are permissible for this design. Combinations of live, dead, and seismic loads in conjunction with thermal stresses have been considered in the design analysis. The design also considered the jet forces that might act on the containment consequent to a pipe severance. Adequate strength has been provided to prevent failure of the containment wall as a result of direct jet impingement, and all pressurized penetrations have been supported with anchors and limit stops to limit pipe movement and prevent failure of the containment. To prevent pipe whip from causing failure of the containment the applicant has restrained the large pressurized pipes within the drywell, or provided energy absorbing structures on the internal walls at strategic locations as described earlier in this report. The possibility of missiles being generated from the failure of flanged joints such as valve bonnets and instrumentation fittings was also considered. The primary containment system design, fabrication, and inspection was in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection B, "Requirements for Class B Vessels" with appropriate addenda and Code Case Interpretations.

Based on our review of this application and similar designs, we conclude that the primary containment design basis is acceptable.

4.1.2 Containment Atmosphere Control

Following a loss-of-coolant accident (LOCA), (a) hydrogen gas could be generated inside the primary containment from a chemical reaction between the fuel rod cladding and steam (metal-water reaction), and (b) both hydrogen and oxygen would be generated as a result of radiolytic decomposition of recirculating coolant solutions. If a sufficient amount of the hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent reaction of hydrogen with oxygen at rates rapid enough to lead to significant over-pressure could lead to failure of the containment to maintain low leakage integrity.

The applicant proposes venting of the containment as the corrective measure if the monitored hydrogen concentration inside the containment shows signs of approaching the lower flammability limits. The proposed venting involves purging containment atmosphere with air and venting in a controlled manner through the standby gas treatment system to the stack.

We have concluded that a hydrogen control system should be provided, in addition to the purging system proposed by the applicant, to keep the hydrogen content within safe limits; i.e., less than 4 volume percent. In its report on the Dresden 3 facility, the ACRS recommended that a reasonable time period be allowed for the design of such a system

and appropriate review by the regulatory staff. Special Report No. 14 submitted for Dresden Unit 3 in response to our letter to Commonwealth Edison Company dated December 22, 1970 provided additional information on post-accident combustible gas control for Dresden Unit 3. This Special Report provides a conceptual design of a flammability control system and a containment atmosphere monitoring system. As was the case for the Dresden Unit 3 facility, installation of a combustible gas control system would require a change in the design of the Pilgrim facility after the construction permit had been issued.

Since we had not considered the problem of combustible gas control during our construction permit review of the Pilgrim application, we evaluated this matter in the light of 10 CFR 50.109 which states that the Commission may require the backfitting of a facility if it finds that such action will provide substantial additional protection which is required for the public health and safety. Our calculations in accordance with AEC Safety Guide No. 7 on other similar plants has indicated that the production of hydrogen is such that purging would be required within about 10 hours following a loss-of-coolant accident. The radiological consequences from such releases using the existing standby gas treatment system calculated by us for another similar plant

indicate that the incremental doses could be significant. Thus the capability to control the hydrogen concentration by measures not requiring release to the environment with the present system would provide a substantial reduction in the total offsite doses that might result from the accident.

We have concluded that the backfitting of the Pilgrim facility in this regard will provide substantial additional protection required for the public health and safety, but that the design and installation of the system need not be completed prior to issuance of an operating license. We believe this action to be consistent with the advice of the ACRS as given on the Dresden 3 facility, i.e., that action should be achieved on a reasonable time basis. We will require the applicant to submit the detailed design and schedule for the installation and testing of a containment atmosphere monitoring system and control system to meet the design basis given in Safety Guide 7.

The primary containment will be provided with an inert atmosphere of nitrogen during reactor power operation, keeping the containment atmosphere oxygen concentration less than 5% by weight, in order to minimize the possibility of the combustion of hydrogen evolved from a zirconium-water reaction during the first few minutes following a loss-of-coolant accident. The inert atmosphere will also extend the time available to cope with the hydrogen evolved from radiolysis of the primary coolant.

We recognize that inerting makes inspection and repair of the primary system more difficult and believe that it is prudent to permit short term personnel access to the drywell for leak inspections during startup/hot standby periods when the primary system is at or near rated operating temperature and pressure. Accordingly, a 24-hour transition period is permitted to inert subsequent to inspections and the placement of the reactor in the Run Mode, and to deinert during operation prior to a shutdown without significantly affecting plant safety.

Containment inerting has been required for all previous boiling water reactor pressure suppression containments. We conclude that the inerting system for the Pilgrim Station provides additional plant safety which outweighs the operational restrictions that may result.

4.1.3 Isolation Valves

The basic function of all primary containment isolation valves is to provide containment integrity between the primary coolant system pressure boundary or the containment atmosphere and the environs in the event of accidents or similar equipment failures. Where necessary the valves are provided with valve operators, and these valves are automatically closed when the sensors detect certain accident or faulted conditions. The consequences of postulated pipe failures both inside and outside the containment have been evaluated. For example, the operational aspects of the main steam line isolation valves for a

steam line break outside the containment are described in the accident analysis given in Section 9.

As a safety system, the isolation valves and their control systems have been reviewed to assure that no single accident or failure can result in a loss of containment integrity. An exception occurs in the case of the instrument lines that connect to the reactor primary coolant system, penetrate the containment and dead-end in instrument transducers located in the reactor building. These 1-inch lines have only two isolation valves, both of which are outside the containment. The inboard valve nearest the containment is a hand-operated globe valve. The second valve, immediately adjacent, is a spring-loaded excess flow check valve. A break in the portion of the instrument line between the containment and the excess flow check valve would result in a blow-down directly to the reactor building.

The applicant has installed orifices in each of these lines inside the primary containment. The orifice size (1/4 inch diameter) selected is sufficiently small that the quantity of coolant that would be discharged from the reactor into the reactor building in the event of a rupture of an instrument line would not result in a failure of the secondary containment, and if the reactor building is isolated, the operation of one standby gas treatment filter train will prevent the pressure in the reactor building from exceeding its design value. Based on our review

of the Pilgrim design, we conclude that the isolation valves and the instrument line orifices for the Pilgrim containment are acceptable.

4.1.4 Testing and Surveillance

During construction, the primary containment vessel was strength-tested at 1.25 times the 56 psig design pressure. The design leak rate is 0.5% of the contained air per day. After installation of all penetrations, integrated leakage tests will be conducted at selected pressures up to 45 psig (the predicted peak accident pressure) to establish reference data for use in evaluating later periodic surveillance tests at 23 psig.

The Technical Specifications require that the primary containment leakage rate measured at a pressure of 45 psig not exceed 1.0 weight percent of the contained air per 24 hours. We used a leakage rate of 1.25 weight percent per day in our calculations of the potential offsite doses for the design basis accidents discussed in Section 9.0 to correct for differences in containment atmosphere; i.e., the containment atmosphere would be initially composed of steam and hot air following an accident, but under test conditions the test medium would be pressurized air or nitrogen at ambient conditions. Considering the differences in mixture composition and temperatures, a correction factor of 0.8 was applied.

Local leakage tests will also be performed to assure that leakage from individual testable penetrations and valves, including the main steam line isolation valves, does not exceed the Technical Specification limits. Based on our review and the experience at other operating plants, we conclude that the Pilgrim testing program will assure maintenance of adequate containment integrity throughout the service life of the plant.

4.2 Secondary Containment

The secondary containment (reactor building) encloses the primary containment vessel and contains the refueling facilities and other equipment provided to support the operation of the reactor. Up to the refueling floor the reactor building is a reinforced concrete structure and, above this, a structural steel frame covered with exterior concrete wall panels and an insulated steel deck roof.

The reactor building is designed as a low pressure low leakage building and provides for the control of any radioactive gases that might be released into the building during a refueling accident or by leakage from the primary containment following a loss-of-coolant accident. During normal operations, the reactor building atmosphere is monitored and exhausted to the environs through the reactor building stack. In the event of the above accidents, the reactor building isolation system would isolate the reactor building stack immediately and the reactor building atmosphere would be processed through the Standby Gas Treatment

System (SGTS) prior to being discharged up the main plant stack. The SGTS consists of two parallel, redundant trains, each with a fan, filters and charcoal beds. Each train is designed to treat a gas flow rate of 4000 cubic feet per minute (cfm). With the reactor building isolated, each train has the necessary capacity to reduce the building pressure and maintain it at a negative pressure of 1/4 inch of water (under neutral wind conditions).

As discussed in Section 9.2 of this report, the applicant has provided a double-bed charcoal filter system to give increased iodine filtration capability. This action is documented in Amendment No. 27. The filters will be tested to demonstrate a removal efficiency for particulates of not less than 99%. The charcoal beds will also be tested to demonstrate that their iodine removal efficiency is not less than 99%. A test program will be conducted before reactor operation and periodically during the life of the plant to demonstrate the design capability and operability of the secondary containment and SGTS.

Based on our review of this and other similar systems, we conclude that the design and testing of the reactor building and SGTS are acceptable.

5.0 EMERGENCY CORE COOLING SYSTEMS (ECCS)

5.1 General

The emergency core cooling systems consist of two high pressure systems (the high pressure coolant injection systems (HPCI) and the auto-depressurization system (ADS)), and two low pressure systems (the low pressure coolant injection system (LPCI) and the core spray system). The emergency core cooling systems for the Pilgrim Station are the same systems, except for flow capacity, as the designs previously reviewed and accepted for Monticello, Quad-Cities, and Vermont Yankee plants. Certain of the systems are similar in design and equipment to the corresponding systems on Millstone 1 and Dresden 2 and 3.

The emergency core cooling systems are designed as Class I (seismic) systems. All piping within these systems is designed and fabricated to the requirements of the Power Piping Code, USAS B31.1.0 - 1967, and the supplementary requirements of the FSAR Appendix A. We have reviewed the piping classification for these systems and the design, fabrication and inspection requirements proposed for each classification and find them acceptable.

5.2 ECCS Objectives

The ECCS subsystems provide emergency core cooling during those postulated accident conditions where it is assumed that mechanical failures occur in the primary coolant system piping resulting in a loss of

coolant from the reactor vessel greater than the available coolant makeup capacity using normal operating equipment. The ECCS subsystems are provided of such number, diversity, reliability, and redundancy that no single failure of ECCS equipment occurring during a loss-of-coolant accident will result in inadequate cooling of the reactor core.

Each of the ECCS subsystems is designed to function over a specific range of primary coolant piping system break sizes. For small breaks in liquid line , up to about 0.10 ft^2 in area, the high pressure coolant injection (HPCI) subsystem is capable of supplying sufficient coolant to depressurize the vessel and cool the core. Delayed initiation of the core spray subsystem and/or the LPCI mode of the RHRS would provide long-term core cooling. For breaks between 0.10 ft^2 and 0.2 ft^2 in area in liquid lines, the depressurizing function of the HPCI and the large volume coolant makeup capability of either the core spray subsystem or the LPCI mode of the RHRS would act in combination to provide effective core cooling. In the event of a loss-of-coolant accident without high pressure coolant injection capability (i.e., the normal feedwater and HPCI are assumed to be unavailable), the ADS would cause the reactor vessel blowdown to occur in a time interval sufficiently short to permit core spray and/or LPCI mode operation with rapid vessel reflooding before excessive fuel clad heating occurs.

For breaks in liquid lines larger than about 0.2 ft^2, depressurization assistance is not required. The core spray subsystem by itself and in conjunction with the LPCI mode of the RHRS are capable of cooling the core independently of the HPCI or ADS for a range of break areas from approximately 0.2 ft^2 up to and including 4.4 ft^2, the latter corresponding to the double-ended break of the largest primary coolant (recirculation) pipe. Both the LPCI mode of the RHRS or core spray subsystem are designed to respond quickly to the larger break sizes with large volumes of coolant water in flooding and spraying modes respectively.

In the case of steam line breaks within the drywell, the ECCS objectives are satisfied more easily for breaks in steam lines than for breaks in liquid lines because the reactor primary system depressurizes more rapidly with less coolant mass loss for steam breaks than for liquid breaks. For example, the HPCI system is capable of providing short term core cooling for steam line break sizes up to about 0.7 ft^2.

5.3 High Pressure Coolant Injection System (HPCI)

The HPCI system provided for Pilgrim is substantially the same as the system provided on Vermont Yankee, and similar except for sizes and capacities to the systems provided on Monticello, Quad-Cities, and Dresden 2 and 3. The HPCI system includes one steam-turbine-driven

pump injecting 4250 gpm of high pressure cooling water through one of the feedwater lines into the reactor vessel. Steam for the turbine is drawn from one of the main steam lines within the drywell and turbine exhaust steam is discharged into the torus water through a submerged pipe. The pump takes suction first from the two condensate storage water tanks with an automatic transfer of suction to the torus water if additional water is required.

5.4 Auto-Depressurization System (ADS)

The ADS system on Pilgrim utilizes four of the Target Rock dual purpose relief and safety valves. The ADS system for Pilgrim is substantially the same as the systems provided on Millstone 1 and Monticello, except that those systems use only three Target Rock valves.

The applicant has modified the original design to include an interlock to prevent automatic actuation of the automatic pressure relief system unless one of the LPCI or core spray pumps is operating. This modification is consistent with the designs approved on the above two plants and satisfies the ACRS concern identified during the Pilgrim construction permit review. We conclude that this matter is resolved.

5.5 Low Pressure Coolant Injection System (LPCIS)

The LPCI mode of the RHR system provides rapid flooding of the reactor vessel in the event of a large break loss-of-coolant accident.

Protection provided by the LPCI mode also extends to a small break, in which the feedwater, control rod drive water pumps, RCIC, and HPCI are all unable to maintain the reactor vessel level and the ADS has operated to lower the reactor vessel pressure so that LPCI and the core spray system start to provide core cooling. The containment spray mode of the RHR system provides spray cooling to the drywell and suppression chamber after the reactor core has been reflooded following a loss-of-coolant accident. The design and equipment for these portions of the RHR system performing these two functions are substantially the same as the subsystems provided on Quad-Cities and Millstone 1, and similar except for capacities to the subsystems provided on Monticello and Vermont Yankee. The major equipment of the RHRS consists of four main system pumps and two heat exchangers for long-term core and containment cooling. The equipment is connected by associated valves and piping and the controls and instrumentation are provided for proper system operation. The main system pumps are rated at a flow of 4800 gpm at 20 psid.

5.6 Core Spray System

The core spray system provides high volume spray to the reactor core in the event of a large break loss-of-coolant accident. It consists of two independent subsystems drawing water from the suppression chamber, and pumping directly into the reactor vessel and onto the core through

the two core spray headers. Each of the core spray pumps are designed to deliver 3600 gpm at 104 psid. System design is based on the assumption that only 1 of the 2 core spray pumps are required to deliver core spray flow. The core spray system provided for Pilgrim is substantially the same as the system provided on Millstone 1 and is similar except for capacity to the systems provided on Monticello, Quad-Cities, Vermont Yankee and Dresden 2 and 3.

5.7 Net Positive Suction Head (NPSH) to RHR and Core Spray Pumps

During the course of the construction permit review on Pilgrim, we questioned whether the RHR and core spray pumps, and their respective systems, were designed to provide an adequate NPSH margin to assure their continued operation following a loss-of-coolant accident. In Amendment 9 to the application, Boston Edison Company furnished an analysis based on preliminary design assumptions showing that a positive NPSH margin would be available following the accident without requiring containment overpressure. The applicant provided further information in Amendment No. 24 with an analysis confirming the final design requirement that a positive NPSH margin be available even if the containment spray were operating following a design basis loss-of-coolant accident (LOCA). We conclude that the equipment provided is adequate to assure sufficient NPSH to the emergency system pumps in the unlikely event of a LOCA.

5.8 Emergency Core Cooling System Performance Evaluation

5.8.1 General

The AEC Regulatory Staff has conducted a general reevaluation of the emergency core cooling systems (ECCS) for light water reactors. Analytical methods and models were the principal areas considered in the review. Experimental results accumulated over the past several years and their applicability to these methods and models were also examined.

Technology associated with analyses of ECCS performance capability has developed substantially in the past five years. As a consequence of this expanding technology, improvements in analytical techniques have been evolving. Some of these improvements were incorporated into the evaluation techniques used for facilities already licensed for operation and for other facilities for which the regulatory review was essentially complete. Recently, additional changes and improvements have been made in these analytical techniques for use in analyses of loss-of-coolant accidents. These changes and improvements are documented in topical reports submitted by the General Electric Company. As a result of these changes and because of the results of some preliminary safety research experiments, we have reevaluated the effectiveness of emergency core cooling systems in light water reactors.

Recent experiments (December 1970) performed by the Aerojet Nuclear Corporation* in support of the Loss of Fluid Test (LOFT) program revealed that certain then-current analytical techniques and assumptions were inadequate to describe certain phases of a blowdown experiment in a small-scale test rig. These experiments, which were part of a series designated as the Semi-scale Blowdown tests, were intended to provide data that could be used to check certain analytical techniques to be used in conjunction with design of the LOFT program. The LOFT program and associated separate effects tests (including the Semi-scale Blowdown experiment) are designed to investigate loss-of-coolant phenomena for pressurized water reactors. We have reviewed the results of the Semi-scale Blowdown tests and have concluded that although they provide useful input to the continuing development of analytical techniques, the results can not be applied directly to power reactors in general, and that specifically they are not relevant to the evaluation of the performance of emergency core cooling systems for boiling water reactors.

On June 19, 1971, the AEC issued an Interim Policy Statement containing interim acceptance criteria for the performance of emergency core cooling systems in light-water nuclear power plants. The June 19, 1971, statement reads in part:

> "The Atomic Energy Commission has recently been reevaluating the theoretical and experimental bases for predicting the performance of emergency core cooling systems; including new

*Formerly the Idaho Nuclear Corporation

information obtained from industry and AEC research programs in this field. As a result of this reevaluation, the interim criteria of Section IV of this Policy Statement have been adopted by the Commission for use in the licensing of light-water power reactors."

Section IV of the Interim Policy Statement states that an acceptable evaluation model for General Electric reactors is given in Appendix A, Part 2 of the Statement. This part of Appendix A states:

"Analysis should be performed for the entire break spectrum up to and including a double-ended severance of the largest pipe of the reactor coolant pressure boundary. The combinations of systems used for analysis should be derived from a failure mode and effects analysis, using the single failure criterion as indicated in Table 2-1 of the GE topical report "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329. The analytical techniques described in NEDO-10329 and its supplement should be used with the following exceptions:

1. During the period of flow coastdown after the minimum critical heat flux ratio at the hot spot is less than one and until the top of the jet pumps uncover, the heat transfer coefficient should be calculated using the D. C. Groeneveld correlation (AECL-3281, equation 5.7).

2. During the period of lower plenum flashing until the core becomes uncovered, the heat transfer coefficient should be calculated using Groeneveld's correlation as in (1) above.

3. The heat transfer coefficients associated with rated core spray flow should correspond to those derived from experimental data, assuming the cladding and channel box emissivity is equal to 0.9.

4. It should be assumed that channel wetting does not occur until 60 seconds following the wetting time calculated using the Yamanouchi analysis.

5. A range of conservatively calculated peaking factors should be studied and the combination selected which results in the most severe thermal transient for the break spectrum and combinations of systems analyzed.

6. The decay heat curve described in the proposed ANS standard, with a 20% allowance for uncertainty, should be used. The fraction of decay heat generated in the hot rod should be considered to be 100% of this value unless a smaller value is justified. The effects of voids on reactivity during a blowdown may be taken into account."

Following publication of the Interim Policy Statement, the applicant and GE performed additional loss-of-coolant calculations using the criteria and assumptions set forth in the Interim Policy Statement, Particularly Appendix A, Part 2 which describes a General Electric Evaluation model acceptable to the Commission.

This report discusses the results of our review of the Pilgrim Station emergency core cooling systems in accordance with the Interim Policy Statement. Topical Report NEDO-10329 (dated April 1971) and supplements thereto and the results of the applicant's evaluation contained in a letter dated July 22, 1971, provided the bases for the review.

5.8.2 Discussion of ECCS Review

The procedures used by GE to analyze the consequences of a LOCA depend upon the particular break size and the location being evaluated. It has been shown by GE that the 'worst-case' situation (i.e., highest cladding temperature) arises for a break in the coolant recirculation

lines because they have the potential for causing the coolant mass loss from the vessel to be more rapid and more extensive than for breaks in other lines in the reactor coolant system that would carry either steam or two phase fluid mixtures. Based upon our review of the GE analyses we have found that the 'worst-case' situation, with regard to assessing the performance of the ECCS, would be for an instantaneous break of a large recirculation outlet line.

For the purposes of analyses, the changing thermal and hydraulic phenomena that are associated with a design basis loss-of-coolant accident (LOCA) may be described in five phases: (a) temperature changes and heat removal during reactor blowdown with associated flow coastdown, (b) achievement of critical heat flux at any point on the fuel rod cladding and associated temperature rise of fuel and clad material, (c) lower plenum flashing causing a temporary resurgence of core flow[1/], (d) temperature rise of fuel and cladding with diminished cooling and complete depressurization, and (e) temperature changes and heat removal during ECCS operation. The analyses for each of these phases of the LOCA originally were performed using calculational models and techniques different from the models and techniques currently used by the applicant

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[1/] This phase occurs when the fluid in the lower plenum reaches a saturation condition resulting in a rapid expansion of the fluid causing a large flow increase in the core region.

(and GE). Since the original analysis, additional information and results of tests related to the performance of the ECCS systems also have become available. As discussed in the following sections, we have met with GE on many occasions to review details of the codes, models, and analyses. In addition, independent checks of certain of these calculation modes using different codes have shown reasonably good agreement with the GE results.

Flow Coastdown

The first phase of the LOCA is the short-term blowdown period during which energy is removed from the core by coolant passing through the core and exiting through the postulated break, causing the reactor coolant system pressure to decrease rapidly. Initially, conditions are nearly the same as during normal operation and nucleate boiling continues undisturbed. During the nucleate boiling regime, a heat transfer coefficient of about 3×10^4 Btu/hr-ft^2°F is calculated to exist during this period of time which is about 5 seconds.

A short time later, the core flow and system pressure decrease sufficiently that nucleate boiling cannot be sustained and the heat transfer rate decreases markedly. In previous analyses a "dry-out" model was used to calculate the time at which the degradation in heat transfer occurs. This model was based on the results of tests in which the flow in a heated test section was stopped simultaneously with initiation of depressurization. Because the flow in a BWR core after a pipe break

is expected to coast down rather than to stop instantly, degraded heat transfer will occur later than would be predicted using the "dry-out" model. In the current analyses, the time at which the departure from the nucleate boiling (DNB) occurs is determined using empirical correlations based on the results of steady-state critical heat flux (CHF) tests.

We requested additional information from GE to confirm the conclusion that the use of the steady-state CHF correlation is appropriate. GE has performed transient tests to demonstrate the validity of their steady-state CHF correlations during transient conditions. Tests were performed in which flow and pressure were reduced separately and concurrently. Comparison of the CHF measured in the tests with the CHF predicted by the steady-state correlation show that the use of the correlation gives results that are conservative.

Flow Stagnation Period

The short-term blowdown phase ends when the coolant flow through the core is assumed to stop as the water level in the downcomer region reaches the inlet of the jet pump. Even though flow actually would continue at a very reduced rate, for conservatism in the analysis, the flow in the core is assumed to stop at this time. During this period of flow stagnation, the heat transfer coefficient used in the analysis

is assumed to be equal to zero. This implies that no heat transfer by conduction or convection from the fuel rods to the coolant occurs and the fuel heats up adiabatically except for the heat loss due to thermal radiation from the fuel surfaces which is assumed to take place during this phase, which exists for about 2 seconds. The GE analysis yields a maximum calculated cladding temperature during this period of somewhat less than 1200°F. Similarly, GE calculates that the pressure in the primary system during this second phase of the postulated LOCA is continuing to decrease because of mass loss through the break, and the depressurization rate during the short-term blowdown regime is of the order of 20 psi/sec. We conclude that the use of a heat transfer coefficient equal to zero for this period is very conservative.

Lower Plenum Flashing

In the third phase of the LOCA, lower plenum flashing occurs. This is a flow phenomenon during the blowdown wherein a sudden transient increase in the core flow begins a few seconds after the core flow has decayed to near zero. The increase in core flow results when the fluid in the lower plenum reaches a saturated condition. At about 10 seconds after the start of the accident the liquid level in the vessel drops below the recirculation line suction nozzles causing the flow out the break to change from a liquid phase to a steam phase and increasing the rate of depressurization of the system (to about 60-70 psi/sec) resulting in additional flow through the core.

The rapid depressurization results in a rapidly changing thermodynamic state of the fluid in the primary system. Because the fluid in the lower plenum beneath the core was initially in a subcooled state (by about 25 Btu/lb), it does not change thermodynamic state during early blowdown as does the rest of the fluid system; however, when the system pressure decreases to the level where this fluid flashes to steam a large increase in steam flow through the core results. This period of the LOCA is called "lower plenum flashing". Calculation of flows, temperatures and pressures during this phase depends on the knowledge of the flashing process, the effect of flow maldistribution, the resistance to flow of a two-phase mixture through the core and jet pump diffusers, and the rate of blowdown through the break.

During this period of increased core flow, GE assumes that nucleate boiling is re-established and that relatively large heat transfer coefficients result. Although nucleate boiling may be re-established, there is insufficient experimental evidence to support this assumption. Therefore, we asked GE to perform the analysis with the assumption that only stable film boiling occurs, with greatly reduced values of heat transfer coefficient, as determined by the Groeneveld correlation, during the period of lower plenum flashing. This assumption is in accord with Appendix A, Part 2 of the Commission's June 19, 1971, Interim Policy Statement on emergency core cooling systems.

Core Heatup

Following the period of lower plenum flashing it is conservatively assumed that no convective cooling occurs. Heat generation, produced by the radioactive decay of the fission products, and thermal radiation among the fuel rods causes the core to heat up. The results presented by the applicant show the calculations of fuel clad temperature in the core for four fuel rod groups. We have reviewed the calculations for this period and conclude that the predicted thermal responses calculated during this phase are conservative.

ECCS Operation

Although the loss of water level or the increase in drywell pressure resulting from a pipe break is sensed immediately and the ECCS is signaled to start, the actual injection of water by the low pressure systems does not occur for about 30 to 40 seconds. This time is required for the diesel generators to start and accept load, the reactor pressure to fall below the ECCS pump discharge pressure and the ECCS pumps to achieve full flow. Water is injected into the reactor through both the LPCI system and the core spray system.

In accordance with the Interim Policy Statement two limiting cases are considered, assuming that a single failure of an active component might result in the operation of both core spray systems only, or in the operation of one core spray system in conjection with half (2 out of 4) of the LPCI pumps.

GE has recalculated the fuel rod temperatures during the design basis LOCA, using a core spray model and heat transfer correlation based on the data from spray cooling tests of electrically-heated full-length fuel assemblies containing rods that were clad with either stainless steel, or zirconium (FLECHT tests). The results of these tests of full-length assemblies are summarized in the report submitted by GE and titled "Effects of Cladding Temperature and Material on ECCS Performance" (NEDO-10179). The purpose of the tests was to demonstrate the effectiveness of top spray and bottom flooding in cooling full-sized BWR assemblies at temperatures and power levels representative of the Pilgrim Station core.

In order to accomplish this purpose nearly 150 tests were run at conditions and with bundles that were as nearly representative of the expected accident conditions and core configurations as possible. The tests did not simulate the blowdown phase of a LOCA, but used the expected conditions at the time of initiation of the low pressure core standby cooling systems as initial conditions for the tests. Most of the tests were run using top spray and bottom flooding concurrently.

GE has developed a heat transfer correlation based on data obtained in the stainless steel bundle tests. This correlation is reported in the GE topical report "Loss-of-Coolant Accident and Emergency Core Cooling

Models for General Electric Boiling Water Reactors" (NEDO-10329). The correlation accounts for the reduced heat transfer in the central pins of an assembly more realistically than previous correlations, predicts the clad temperatures more accurately, and forms the basis for applying the stainless steel test results to Zircaloy bundles. We have reviewed the GE correlation and its application, and a core spray model developed independently by the Aerojet Nuclear Corporation (ANC). In both models, heat transfer by thermal radiation from the rods is the controlling phenomenon in cooling of the rods, and heat transfer by convection plays only a small part.

The convection portion of the heat transfer is inferred from the results of the tests on stainless steel bundles by subtracting the thermal radiation heat flux from the total heat flux as measured in the tests. The thermal radiation flux is calculated using the geometrical view factors and experimentally determined emissivity. In the GE analysis a value of 0.6 was used for the emissivity of both the stainless steel rods and channel box. Subsequent tests made to measure the emissivity of the stainless steel bundles indicated that a value of 0.9 would be more appropriate. Since the use of the larger value of test bundle emissivity results in smaller values of the empirical convective heat transfer coefficient, we have required GE to perform the analysis of the accident using convective heat transfer coefficients derived using the larger

value of emissivity. This matter is treated in accordance with Appendix A, Part 2 of the Commission's Interim Policy Statement on emergency core cooling systems.

Since the cooling is mainly by thermal radiation from the fuel to the channel box, which is the outer container of the fuel bundle, the peak fuel clad temperatures reached are strongly influenced by the temperature of the Zircaloy channel. Until the channel is wetted by action of the spray, it is at a high temperature and is a relatively poor heat sink. GE and ANC use different heat transfer coefficients derived from the same data to calculate the cooling of the channel prior to quenching. GE uses a correlation based on theoretical analysis to calculate the time to wet, or quench. ANC estimates a longer quench time from available data. In order to assure that the channel quench times are conservatively calculated, we have required the calculation of the fuel rod temperatures using a channel quench time which is 60 seconds longer than that determined by the theoretically-based (Yamanouchi) correlation. This requirement is in accordance with Appendix A, Part 2 of the Commission's Interim Policy Statement on emergency core cooling systems.

The results of the applicant's reevaluation of the ECCS performance also include the effects of peaking factor selection and the use of the decay heat data as described in Appendix A, Part 2 of the Commission's Interim Acceptance Criteria for emergency core cooling systems.

The peak cladding temperature calculation assumes that cooling by spray action persists until the accumulation of water from two core spray pumps is sufficient to terminate the temperature transient by flooding action. The assumption of only two core spray pumps results from assuming a single failure of an active component. In this case, the assumed failure is the LPCIS. All other single failures of active compoents, e.g., one core spray pump and two LPCI pumps, being available for cooling the core, result in lower calculated temperatures.

The applicant and GE have performed these analyses for the entire pipe break spectrum, up to and including a double-ended severance of the largest pipe of the reactor coolant pressure boundary. In the limiting case of a postulated double-ended break of a primary coolant system recirculation loop pipe with the simultaneous failure of the LPCI system, the calculated maximum fuel clad temperature is 2200°F, using the AEC evaluation model described in Appendix A, Part 2 of the Interim Policy Statement. The calculated maximum clad temperature for smaller breaks of about 0.1 ft^2 is 1980°F. The cladding-water chemical reaction is calculated to be less than 0.12% of the total amount of cladding in the reactor for all break sizes.

5.8.3 Conclusion

We conclude that the design of the Pilgrim emergency core cooling system is acceptable based on the analysis which shows that the consequences

of the loss-of-coolant accident are such that (a) the calculated maximum fuel rod cladding temperature does not exceed 2300°F, (b) the amount of fuel rod cladding that reacts chemically with water or steam does not exceed 1% of the total amount of cladding in the reactor, (c) the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and before the cladding is so embrittled as to fail during or after quenching, and (d) the core temperature is reduced and decay heat is removed for an extended period of time.

Based on the above, we conclude that the emergency core cooling system meets the requirements of Criterion 35, Emergency Core Cooling, of the General Design Criteria published in the Federal Register on February 20, 1971, in that it is capable of transferring heat from the reactor core following any loss-of-coolant accident at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

6.0 INSTRUMENTATION, CONTROL, AND ELECTRICAL POWER SYSTEMS

6.1 General

Our review of the Pilgrim Station instrumentation, control, and electrical power systems was directed at those subsystems and features which are related to the safety of the plant. Particular attention was given to the review of the reactor protection system, the initiating and control systems for the vital isolation functions, the emergency core

cooling systems, and the emergency power portions of the station auxiliary electric power system. The Commission's General Design Criteria, as published in the Federal Register on February 20, 1971, and the proposed IEEE criteria for Nuclear Power Plant Protection Systems (IEEE 279) served, where applicable, as the bases for evaluating the adequacy of the designs. The reactor protection system, and the instrumentation that initiates and controls the engineered safety features are substantially the same as those in the Monticello Nuclear Power Station, Docket No. 50-263 and are acceptable. Certain features that were not included in or differed significantly from the Monticello plant were reviewed separately. These features are discussed in the following sections.

6.2 Anticipated Transients Without Scram (ATWS)

As further confirmation of the adequacy of design, we and the ACRS have asked the reactor supplier, General Electric, to study means of preventing common mode failures from negating scram action and design features to mitigate the consequences of failures to scram during anticipated transients. General Electric has submitted the results of these studies in two topical reports, NEDO-10189, "An Analysis of the Functional Common Mode Failures in GE BWR Protection and Control Instrumentation" dated July 1970 (submitted October 26, 1970), and NEDO-10349, "Analysis of Anticipated Transients Without Scram" dated

March 1971 (submitted May 4, 1971). Those reports are now under review by the regulatory staff. We and the ACRS believe it is desirable to expedite these studies, and implement such design modifications as are found to improve the safety of the plant significantly. This matter is common to all recently licensed water reactor plants. We have concluded that this study need not be completed prior to issuance of an operating license for Pilgrim.

6.3 Plant Protection System Testability

We have reviewed the reactor protection system and the instrumentation that initiates and controls the engineered safety features to determine that they can be tested during operation of the plant. We have reviewed the schematic diagrams to determine that the entire circuit, from sensors to activated component can be tested during reactor operation. Although an integrated system test cannot be performed during reactor operation, individual components or portions of the system can be activated in a manner that adequately tests the action required in the event of an accident. During our review we found that several subsystems (steam-line leak detection temperature channels, safety relief valve bellows monitor pressure switch, turbine stop valve closure channels, scram discharge volume level channels, primary containment isolation channels, and instrument sensing lines) could not be unambiguously tested. During the course of our review, the applicant has either modified these subsystems or has provided detailed test procedures so that each subsystem can be fully tested. We conclude that acceptable means of testing

the reactor protection system and engineered safety feature actuation circuits during reactor operation have been provided.

6.4 Incident and Accident Surveillance Instrumentation

The BWR reactor protection and engineered safety feature instrumentation channels generally do not provide a continuous quantitative readout in the control room of the parameters being monitored. The neutron monitoring and main steam line radiation monitoring and main steam line radiation monitoring systems are exceptions. The other vital parameters, however, are monitored by instrument channels associated with control systems. Certain information readout channels are required by the operator to assess plant conditions subsequent to an accident in order that he may determine whether to intervene in the operation of the Auto-Depressurization System or to initiate the Containment Spray System. Certain other readout information also may be needed during station operation to aid in avoiding or mitigating the consequences of other abnormal operational occurrences. For these reasons, we have required that the Technical Specifications specify that a sufficient number of these readout channels must be operative as a condition for reactor operation. We conclude that sufficient information necessary to initiate and control the engineered safety features will be available to the operator.

6.5 Independence of Redundant Plant Protection System Channels

One of the plant safety design bases requires that, "There shall be sufficient electrical and physical separation between channels and between logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly." Our review raised questions concerning the implementation of this design basis in three general categories. The first category concerned connection of redundant protection channels to single switches and terminal boards in control room panels. The second category concerned installation of redundant protection system switches on control room panels within a few inches of each other and with their wiring bundled and routed together. The third category concerned the installation of redundant protective system instruments on the same rack outside the control room, connected to the same sensing lines, and located in a potentially hostile environment.

In response to our questions, the applicant reviewed the circuitry and arrangement of common devices in the reactor protection system and redundant engineered safety features control systems to verify that no single failure could disable a safety function. Where a single switch accomplishes redundant functions, barriers are installed between switch sections to preclude the possibility of inadvertent

shorting between adjacent sections. Barriers are installed between adjacent terminal points where potential for shorting exists, and between redundant and proximate switches. Wiring for redundant safety channels within the panels are protected against common events by spatial separation, barriers, or enclosure in conduits. The applicant further established criteria for physical separation and protection against concurrent failure of functionally independent safety systems by single credible events such as missiles, fires, and flooding.

We conclude that the applicant has now provided adequate independence of redundant plant protection system channels against single failures and events.

6.6 Electrical Power Systems

6.6.1 Offsite Power

The Pilgrim Station will be interconnected to the Boston Edison Company system through 345 kV circuits. Power from the unit's generator is fed via a single circuit containing the main step-up transformer to the 345 kV switchyard. The 345 kV switchyard is arranged in a ring bus configuration. Two 345 kV transmission circuits emanate from the switchyard sharing common towers for a distance of about 8 miles. Each of these circuits is capable of carrying full station output and of supplying the startup transformer.

Either of the 345 kV circuits through the switchyard and startup transformer represents the primary source of offsite power to the safe shutdown and engineered safety feature loads. During normal operation, the output of the unit generator is connected through the main step-up transformer. In tripping of the unit generator, auxiliary power for the plant is automatically connected by fast transfer to the start-up transformer. The start-up transformer is also used as a source of power for startup, shutdown, and post-shutdown requirements.

An additional source of offsite power is supplied from a 23 kV circuit through a shutdown transformer to the onsite emergency distribution system. The normal source of power to this 23 kV circuit is of sufficient capacity to supply the area loads and the required engineered safety feature loads of the Pilgrim Station. This 23 kV circuit, however, crosses under the 345 kV lines in two locations, once adjacent to the station and again about three miles from the station. This circuit is mounted on wooden poles which run parallel to public and private roads to the site and the right-of-way parallels that of the 345 kV circuits for a total distance of about 1.2 miles. During the course of our review the applicant agreed to install the 23 kV circuit underground for that portion of its route for which the failure of a 345 kV tower or line might result in a simultaneous loss of the 23 kV circuit. We consider this design change acceptable and responsive to the concern of the ACRS in this matter.

The circuit breakers and the controls for each of these redundant sources of offsite power are physically and electrically independent of each other such that single failures should affect only one of these sources. The stability of the grid has been analyzed and the applicant has reported that the loss of the Pilgrim Station would not affect the availability of offsite power to this station.

On the basis of our review, we conclude that the design of the offsite power system meets the requirements of General Design Criterion 17 of the General Design Criteria as published in the Federal Register on February 20, 1971, and is acceptable.

6.6.2 Onsite Emergency Power

The design of the onsite emergency power system utilizes the split bus concept except as noted and discussed below. The redundant engineered safety feature equipment trains are divided between two 4.16 kV emergency buses such that either one will supply minimum safety requirements. One diesel-generator is connected to each of the 4.16 kV emergency buses. Two stepdown transformers connected to the 4.16 kV emergency buses provide power to two of the three 480V emergency buses.

The two emergency a.c. power systems are independent, except that one 480V bus which supplies power to the LPCI system injection and containment isolation valves can be connected to either of the redundant

and otherwise independent buses. In addition, this bus also supplies power to certain equipment not essential to safety; for example: the turbine turning gear motor and turning gear oil pump, the fire protection system, the 125V d.c. backup battery charger and one air compressor. Normally, this swing bus will be fed from only one of the source 480V buses. However, if power on the source bus should be lost, the swing bus would be transferred automatically to the other bus. The automatic transfer feature includes redundant interlocks to prevent the independent buses from being connected together. Because all of the equipment connected to the swing bus represents a substantial electrical load, we questioned whether its transfer in one block might jeopardize both emergency a.c. power supplies. Therefore, the applicant has designed the system to automatically disconnect or shed all nonessential loads on this bus during the course of effecting a transfer. We have reviewed this design and conclude that it is acceptable.

In the event voltage is lost on either 4.16 kV emergency bus, the diesel-generator attached to that bus will be started within 10 seconds. A diesel-generator can be fully loaded within 30 seconds after a start signal. Should either diesel-generator fail to restore voltage to its respective emergency bus, this bus will transfer to the shutdown transformer (23 kV offsite power source) after a suitable time delay. As a further protective measure both diesel-generators

are automatically started on a loss-of-coolant accident signal, but are not connected to the emergency buses if the normal auxiliary power to these buses is available.

The two redundant diesel-generator sets are each housed in reinforced concrete Class I structures such that there is complete physical independence between units. Auxiliary systems for these machines are redundant, independent and also housed in Class I structures. Sufficient fuel supply is available to permit the operation of each diesel-generator, under LOCA conditions, for seven days.

Three d.c. systems are provided. One system consists of two redundant 125 volt batteries, each with its own charger and distribution system. Further, each is located in a separate Class I room and mounted on Class I racks. These rooms share a common ventilation duct serviced by two redundant fans. The batteries are sized for eight hours of operation. Redundant emergency loads are divided between two distribution panels with a battery assigned to each panel. This d.c. distribution system, like the a.c. onsite system, is a split bus design except that one distribution panel can be supplied from either battery source. Since this load center is complementary to the operation of the above 480V a.c. swing bus, we have accepted it on the same basis; i.e., loss of either battery (power) source will not cause loss of the

other side of this panel, and loss of the panel will not cause loss of either battery (power) source. The remaining d.c. systems consist of a single 250V battery system, and two redundant 24V batteries, each with its own redundant charger. The 250V battery supplies power primarily to the conventional station emergency equipment and the HPCI system. The two 24V batteries power the source and intermediate range neutron monitoring systems and selected process radiation monitoring systems. The 24V batteries are mounted in separate rooms and the 250V battery shares a room with one of the 125V batteries.

We conclude that the onsite power systems have sufficient capacity, independence, redundancy and testability to perform their safety function even in the event of a single failure and are acceptable.

6.7 Seismic, Radiation and Accident Environment Testing

The seismic design bases for the reactor protection system, the engineered safety feature systems, and the emergency power system require that they withstand the loading forces and perform their required functions in the event of the operational basis earthquake or the design basis earthquake (horizontal accelerations of 0.08g and 0.15g, respectively). The equipment and instruments in these systems are designed to perform their functions during these earthquakes and will be qualified to this requirement by analytical techniques and/or vibration testing techniques. The applicant has described his seismic testing program and significant test

results to date. We conclude that the applicant's program for qualifying this equipment and instrumentation to the established criteria is acceptable.

Relative to radiation damage of safety related equipment, the applicant has stated that his design criterion is, "Safety system equipment shall be capable of withstanding without loss of function the potential effects of the total integrated radiation dose from normal operation during the expected lifetime of the station plus a one-time dose from an accident exposure." The applicant has estimated potential exposures of this equipment over the normal lifetime and resulting from a design basis loss-of coolant accident. He reviewed the exposed equipment that must remain functional during this accident, including the materials (i.e., seals, insulation, lubricants) that are most likely to be subject to radiation damage, and concluded that in all cases the equipment meets the above design criterion. This conclusion is based on radiation damage tests on materials and components. We conclude that the applicant's program for radiation testing of safety related equipment is acceptable.

Safety-related equipment inside the primary containment that must operate in an accident environment is limited to isolation valves, recirculation line valves, auto-relief valves and their associated electrical penetrations and cooling. These valve operators, cables,

and penetrations are designed to remain operable for periods in excess of their required usage under the design basis accident conditions. Qualification tests have been or will be performed on the valve operators, power and control cables, and electrical penetrations to assure the required capability. The test data submitted indicate that all safety-related equipment inside the primary containment can remain operable for the length of time needed to complete its safety function. We conclude that the accident environment testing of this equipment is acceptable.

7.0 AUXILIARY SYSTEMS

7.1 General

The auxiliary systems are described in Section 10 of the FSAR. These process systems normally provide plant services auxiliary to the production of power. In the course of our review, we have directed our attention to the safety related objectives of the respective systems and the manner in which these objectives are achieved. We have reviewed the safety related auxiliary systems for redundancy, independence, and physical separation; and for those criteria that establish the quality of the systems. The latter review considered the appropriateness of the seismic design classification, and the use of suitable codes, standards and specifications for the design, fabrication and inspection of the piping and other components within each system. The safety related items that received special attention in the course of our review are discussed in the following sections.

7.2 Fuel Handling and Storage

Fuel handling and storage facilities are provided for storage and transfer of new and spent fuel. New fuel is stored dry in racks which are spaced to preclude attaining criticality. Spent fuel is stored underwater in the spent fuel storage pool which is located adjacent to the reactor. During refueling, the drywell and reactor vessel heads are removed and the cavity over the reactor is filled with water. Spent fuel is transferred underwater to the fuel pool. Subsequently, spent fuel is transferred to a fuel shipping cask which is submerged in the fuel pool. After loading, the cask is removed from the pool and shipped to a fuel reprocessing plant. As with previous BWR designs, we have considered the capability of the fuel pool to withstand an inadvertent dropping of the fuel shipping cask without causing pool damage that might result in a sudden loss of water. Failure of the reactor building crane or handling slings at the most unfavorable moment could result in the cask falling a distance of about 45 feet (including 38 feet of water) before striking the bottom of the spent fuel pool. Such an impact could result in a loss of the pool water; however, the applicant will provide an energy absorbing system on the floor of the fuel pool in the cask handling area. Analysis by the applicant has shown that with the energy absorber in place, damage to the pool floor will not result in a leakage rate greater than the pool make-up capacity. We conclude that the applicant's method of preventing excessive loss of fuel pool water is acceptable.

7.3 Reactor Building Closed Cooling Water (RBCCW) System

The safety objectives of the RBCCW system are to provide cooling to the pumps of the Core Standby Cooling Systems and their respective building compartments, and to provide a heat sink for the RHR heat exchangers. The system consists of two independent closed loops which may be interconnected through two 12-inch crossties. The valves in these crossties are normally closed.

The applicant stated that the safety-related portions of this system will be designed as a Class I seismic system with respect to their ability to retain integrity and prevent loss of water during and after the operating basis earthquake and design basis earthquake. This system also provides for continued operation of the containment drywell coolers following a seismic event with the primary coolant system remaining intact.

On the basis of the independence and redundancy provided by the two loops in this system, and the use of appropriate seismic design criteria, and codes and standards for the piping system and its major components, we find the RBCCW system acceptable.

7.4 Salt Water Service System (SWSS)

The salt water service system consists of two redundant loops normally interconnected at the pump discharge. On loss of a.c. power the loops are automatically isolated by redundant valves. This system supplies

cooling water to the heat exchangers for the reactor building closed cooling water system (RBCCW) and the turbine building closed cooling water system (TBCCW). These four heat exchangers are located in the northwest end of the reactor building within connecting rooms (3'0" floor elevation). Our review raised the concern that a major pipe break in the salt water service system within this compartment might result in a total loss of both the RBCCW and TBCCW systems due to flooding.

The applicant responded to this concern by upgrading the entire salt water service system to a Class I seismic design, by adding a water-tight barrier between the two rooms (and loops), and by providing for rapid draining of either compartment with suitable instrumentation for early detection of flooding conditions.

We have concluded that the salt water service system is acceptable on the basis of the redundancy and independence provided by the two loops in this system; the use of appropriate seismic design criteria, and codes and standards for the piping system and its components; and the protection provided against compartment flooding in the respective loop equipment rooms.

8.0 RADIOACTIVE WASTE SYSTEMS

8.1 General

The radioactive waste systems of Pilgrim Station are designed to collect and process the liquid, gaseous, and solid wastes which are

byproducts of station operation and which might contain radioactive materials. If feasible, which is the case for most liquid wastes, the wastes are treated and returned to plant use. Other liquid wastes are temporarily stored to provide time for radioactive decay, and ultimately discharged at a controlled rate into the plant cooling water. The plant cooling water provides for the dilution of these radioactive wastes to concentration limits below the limits permissible by 10 CFR Part 20. The presently installed gaseous radwaste system also provides a delayed release to reduce radioactivity in the gaseous effluent by radioactive decay. The elevated release point, or main stack, moreover, serves to disperse and diffuse these gaseous wastes. The solid wastes, primarily filter materials and demineralizer resins are packaged, stored and eventually shipped offsite to a Commission approved disposal facility. All releases of radioactive materials from the radioactive waste systems will be limited to the quantities permissible by the Technical Specifications. In addition to limiting releases, the applicant will also conduct an environmental radiation monitoring program in order to assess the effects, if any, on the environment.

On June 9, 1971, the Commission published in the Federal Register (36 F.R. 11113) a proposed amendment to its regulation 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to provide

numerical guidance for establishing the equipment design objectives and Technical Specification requirements for light-water-cooled nuclear power plants consistent with keeping radioactivity in effluents and resultant radiation exposures to levels which are as low as practicable. The applicant and the Commission's staff are currently evaluating the capability of the Pilgrim radioactive waste systems to comply with the recent proposed guidelines. It appears likely that the liquid and gaseous radwaste systems will require modifications to meet these proposed guidelines. The status of each system is described in the respective subsections below.

8.2 Liquid Radwaste System

The liquid radwaste system consists of the tanks, piping, pumps, process equipment, and instrumentation necessary to collect, process, store, analyze, monitor, and discharge any potentially radioactive wastes from the plant. The major volumes of liquid wastes are collected in two process streams according to chemical purity. High-purity liquid wastes are collected from the process equipment drains throughout the plant. These high-purity liquid wastes are treated by filtering and demineralization and then returned to the condensate storage tank for reuse in the station. The impurities and radioactivity removed by the filters and demineralizer resin beds are ultimately shipped offsite as solid radioactive wastes. The low-purity liquid wastes are collected from the floor drains and sumps, the laboratory drains, and the chemical

wastes from regenerating the condensate demineralizers. These low-purity wastes are presently filtered, passed to a monitor tank where their radioactivity concentration is determined, and then discharged to the circulating water discharge canal. The data given in Table 9.1.1.1 (of Amendment No. 16) indicates that the radioactivity concentrations discharged to the canal with the existing system will annually average less than about 4% of the permissible concentrations allowable by 10 CFR Part 20, Appendix B, Table II, on an identifiable isotope basis.

On November 2, 1970, we advised the applicant that he should install a waste evaporator in the liquid radwaste system to reduce the radioactivity that would otherwise be routinely discharged in the plant effluent. The applicant has indicated to us that he expects this radwaste concentrator to be made operable during the summer of 1972. The applicant will install this waste concentrator in the low-purity liquid waste stream just upstream of the three 20,000 gallon monitor tanks. This arrangement will increase the holdup time for these liquids from 8 hours to 20 days, resulting in additional decay in the total radioactivity released. The distillate from the waste concentrator will be pumped to the high-purity liquid waste stream and then into the condensate storage tank for reuse in the plant.

The applicant predicts that the radioactivity concentrations discharged to the canal after the concentrator becomes operational will annually

average less than 1.2% of the maximum permissible concentrations allowable by 10 CFR Part 20, Appendix B, Table II, on an identifiable isotope basis.

In response to the Commission's recently proposed numerical guidelines described above, the applicant has informed us that he is currently studying the technical feasibility of incorporating a facility for solidification and packaging of the waste concentrator bottoms as a means of reducing still further the total radioactivity discharged into the plant liquid effluent. We will continue to review and discuss this matter with the applicant, and if modifications are feasible that provide a significant reduction in the radioactivity discharged we will take steps consistent with the Commission's policy of implementing the numerical guidelines to require its installation on a timely basis.

A third liquid waste process stream results from the backwash liquids used to clean-up the various filters and powdex demineralizers incorporated in the several low-flow water quality systems throughout the plant; i.e., reactor water clean-up system, fuel pool cleanup system, etc. A flocculating agent is mixed with these liquid wastes and the solid impurities are removed by a disposable filter bag. The liquids are returned to the high-purity liquid waste stream for processing and ultimate reuse in the plant. The filter bag is shipped offsite as solid radioactive wastes.

In summary, we have reviewed the liquid radwaste system, including the forthcoming addition of the waste concentrator, and find the system acceptable under the Commission's current Rules and Regulations for the release of radioactive materials in nuclear power plant effluents. Both the applicant and the Commission's staff are evaluating the capability of the liquid radwaste system to satisfy the proposed numerical guidelines for radioactive releases. The applicant's program was described in his letter, dated July 12, 1971, which also stated that he had initiated further engineering studies of equipment and structures that might be required to meet these proposed numerical guidelines. We conclude that the applicant's system is acceptable and that the proposed program for further action is appropriate.

8.3 Gaseous Radwaste System

The gaseous radwaste system provides for disposal of potentially radioactive gases through the main plant stack. The system includes provisions for hold-up or delayed release to benefit from radioactivity decay, and appropriate monitoring, dilution, and automatic shutoff facilities. Normal station operation produces two sources of gaseous and airborne particulate activity discharged to this stack; offgas from the main condenser steam jet air ejectors and exhaust from the main steam turbine gland seal system. During plant startup, the condenser mechanical vacuum pump discharge is combined with the exhaust from the main steam turbine gland seal system and routed to the stack. The standby gas system also discharges to the stack through its own exhaust duct.

During normal operation the steam jet air ejector off-gas is the major contributor to the activity in the plant's off-gas release. The non-condensable gases from the main condenser consist primarily of nonradioactive hydrogen and oxygen formed in the reactor by radiolytic decomposition of water, air in-leakage to the turbine-condenser, radioactive activation products, and some radioactive fission gases, i.e., the inert xenon and krypton gases.

The present air-ejector off-gas system consists of a 30-minute holdup line, high efficiency particulate filters, dilution fans, and the plant main stack. The system piping, from the air ejector outlet to the stack inlet, is designed for a pressure of 350 psi to contain a possible explosion resulting from the hydrogen and oxygen present. The off-gas filter system which is located in the base of the stack, consists of two parallel sets of full-flow, high efficiency, particulate filters. The spare set of filters provides backup and assures availability of filtration. The dilution fans are also located in the base of the plant stack and supply dilution air to reduce the hydrogen concentration in the stack and maintain suitable exit velocities at the top of the stack.

Continuous radiation monitoring of the steam jet off-gas system provides the means to demonstrate compliance with the stack release rate limits. Radiation monitors are located before and after the holdup system. If

the radiation monitors located ahead of the holdup system detect a radioact-/ity level equivalent to the instantaneous stack release limits, they will activate an alarm and timer in the control room. If the high radiation levels persist for longer than 15 minutes the off-gas system is automatically isolated from the stack. Since continued power operation of the plant requires a vacuum in the main condenser, the plant operator would be required to reduce the radioactivity of the gases entering the holdup system consistent with the instantaneous release limits or the plant operator would be required to shut down the plant.

The applicant is required to develop and following operating procedures for the control of effluents and to maintain and use the equipment that has been installed in the radioactive waste systems to keep radioactive materials in effluents as low as practicable. It is expected that compliance with these requirements will keep annual average releases of radioactive materials in effluents at small percentages of the limits set forth in Section 20.106 of 10 CFR Part 20. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual conditions which may temporarily result in releases higher than such small percentages, but still within the limits specified in Section 20.106, and as further specified in the Technical Specifications. The applicant

has stated that he "plans to operate the station so as to maintain the release of radioactive effluents to the environs as low as practicable".

In addition to the requirement that the applicant maintain and use the radioactive waste system equipment, and to develop the following operating procedures for such use to keep releases as low as practicable, the Technical Specifications also specify a limit on maximum releases from the main stack, which reflect the limits of Section 20.106 of 10 CFR Part 20.

Specifically, the limit is not likely to cause any individual to receive a dose to the whole body in any period of one calendar year in excess of 0.5 rem, assuming the plant operates at its maximum release limit. This basis permits a calculated maximum stack release rate of 0.36 Ci/sec. The applicant has agreed to an additional Technical Specification limitation that the average release rate from the main stack will not exceed 0.10 Ci/sec.

We find the limitations described above are acceptable for the interim period until additional gaseous radioactive waste equipment, discussed below, is designed, procured and installed to assure that the operation of the Pilgrim Station conforms to the proposed numerical guidelines published by the Commission June 9, 1971. These interim limitations require effluent releases to be limited to values that are below 10 CFR 20 limits as noted by the ACRS in its report.

In Amendment 29, submitted May 15, 1971, the applicant indicated that he planned to modify the present off-gas holdup system to significantly increase the 30-minute holdup time by installing a hydrogen recombiner and other system modifications. In Amendment 30, submitted June 21, 1971, the applicant indicated that the design of the hydrogen recombiner and gas holdup system would have the objective of limiting the exposure of any offsite individual to 5 millirem per year.

At the present time the design of the system modifications are not complete, and we have requested additional information concerning the proposed modification to allow us to determine the capability of the system to meet the provisions of the Commission's June 9, 1971, proposed numerical guidelines.

By letter dated July 12, 1971, Boston Edison stated that it was reviewing the effects of the guidelines in the proposed amendment to 10 CFR Part 50 on the design of its proposed systems for reducing both gaseous and liquid radioactive effluents. A report on these systems will be provided for our review in the fall of 1971. Operation of the equipment is expected on or before mid-1974. We will pursue on a timely basis the design, procurement, and startup of the additional equipment necessary to make the Pilgrim gaseous radwaste system conform to the Commission's proposed numerical guidelines.

8.4 Solid Radwaste System

The solid radwaste system provides for collecting, sorting, and packaging for offsite shipment the solid radioactive wastes resulting from plant operation. Presently, the bulk of the solid radioactivity wastes consists of spent ion exchanger resins. The remaining solid wastes consist of filter sludges, air filters and miscellaneous paper and rags.

Ion exchange resins and filter sludges are dewatered and placed in shielded casks. Dry wastes are compacted in drums. No solid wastes will be stored permanently at the Pilgrim site. All solid wastes ultimately will be shipped offsite for disposal at an AEC licensed disposal site.

As described under the liquid radwaste system, the applicant is presently studying means of solidifying and packaging the waste concentrator bottoms. Should this study develop a feasible system, these packages of radioactive solid wastes will also be shipped offsite for disposal at an AEC licensed disposal site.

9.0 ACCIDENT ANALYSES

9.1 General

We have evaluated the response of the facility to various anticipated operating transients. The events that characterize abnormal operating

transients have been described by the applicant in Section 14 of the FSAR and include such events as process system control malfunctions, inadvertent control rod withdrawal, turbine trip, loss of electrical load and variations in operating parameters. We have reviewed the results of the applicant's analyses of these events and conclude that the design of the facility, including the protection and control systems, is such that the occurrence of such transients would not result in damage either to the fuel or to the primary coolant boundary. Consequently, the occurrence of these abnormal transients would not lead to a significant release of fission products to the environs.

We also have evaluated a broad spectrum of accidents that might result from postulated failures of equipment, or maloperation. We have selected four highly unlikely accidents (design basis accidents) that are representative of the spectrum of types and physical locations of postulated causes and that involve the various engineered safety feature systems provided. The calculated potential consequences of the design basis accidents exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents to assess the capability of the engineered safety features to control the possible escape of fission products from the facility. The design basis accidents analyzed were: (1) control-rod-drop, (2) refueling, (3) steam-line-break, and (4) loss-of-coolant accidents.

Our evaluation of these accidents showed that the calculated doses resulting from these postulated accidents are well within the 10 CFR Part 100 guideline values. The results of our analyses are summarized in Table 9.0, and the analytical method and assumptions used in each analysis are described in the following sections.

9.2 Loss-of-Coolant Accident Inside Drywell

In calculating the potential consequences of the postulated loss-of-coolant accident, to provide a conservative assessment we have arbitrarily assumed that in spite of the operation of the emergency core cooling system, large amounts of fission products would be released from the reactor fuel. The fractions of the total core fission product inventory we assumed to be released from the core are those given in AEC Safety Guide 3 (published 11/2/70), i.e., 100% of the noble gases, and 50% of the halogens. In addition, 50% of the halogens released from the core are assumed to plate out onto internal surfaces of the containment building or onto internal components. The primary containment was assumed to leak at a constant rate of 1.25 percent of

the containment volume per day for the duration of the accident without consideration of the mitigating effects of decreasing pressure during the post-accident interval.

We have assumed a 95% halogen removal efficiency for the double-bed charcoal absorbers of the standby gas treatment (SGTS) in the secondary containment building. In our analysis, we adopted the conservative assumption that leakage from the drywell goes directly to the standby gas treatment system without mixing in the reactor building and then passes through the SGTS to the environment via the stack, exhausting at 400 feet MSL.

The highest calculated two-hour doses at the exclusion zone boundary occur at the site boundary 330 meters northwest of the main stack. We calculate the doses for a two-hour exposure at this point to be 110 rem to the thyroid and 13 rem whole body. Fumigation conditions with Pasquill Type F stability and a 2 meter per second wind speed were assumed for this analysis.

In calculating the doses at the Low Population Zone distance, we assumed a zero effective release height compatible with assuming downwash over Manamet Hill, which reaches a height of 395 feet MSL. For the first eight hours, we used a dilution factor based on Pasquill Type F and a two meter per second wind speed, using the centerline

concentration. This condition was assumed to continue for 24 hours; but from 8 to 24 hours, the plume was assumed to spread uniformly in a 22-1/2 degree sector. For the next three days, the wind was assumed to continue blowing into the same sector, but variations in stability and higher wind velocities were also assumed. After four days, the same diffusion conditions were used as in the period from one to four days, but the wind was assumed to remain in the sector only 1/3 of the time.

Based on these assumptions, we calculate that the maximum radiation dose during the entire course of the accident at the Low Population Zone distance would be 70 rem to the thyroid and 3 rem whole body.

9.3 Refueling Accident

In evaluating the postulated refueling accident, we assumed that during fueling handling operations, a fuel bundle falls with sufficient force to damage (perforate the cladding) 111 fuel rods. We also assumed that 20% of the noble gases and 10% of the halogens from the damaged rods are released to the refueling pool water. Ninety percent of the halogens released from the perforated fuel rods are assumed to remain in the refueling pool water. The remaining airborne fission products (20% of the noble gases and 1% of the halogens contained in the fuel) within the building are assumed to pass through the standby gas treatment system (with a charcoal adsorber iodine removal efficiency

of 95%) and be discharged from the main stack over a two-hour period. It is assumed that the accident occurs 24 hours after shutdown. The meteorological conditions assumed are the same as described above for a loss-of-coolant accident.

The maximum calculated radiation dose for exposure for two hours at the site boundary 330 meters NW of the main stack is 20 rem to the thyroid and less than 2 rem whole body. The calculated dose for exposure during the course of the accident at the Low Population Zone distance is 2 rem thyroid and less than 1 rem whole body.

9.4 Control-Rod-Drop Accident

For the postulated control-rod-drop accident it is assumed that a bottom-entry control rod has been fully inserted and has stuck in this position unknown to the reactor operator. It is then assumed that the drive becomes uncoupled and withdrawn from the rod. Subsequently, it is assumed that the rod falls from the core inserting an amount of reactivity corresponding to the worth of the rod.

Hot standby is the worst operating condition at which the accident could happen both because a higher energy release is calculated for this condition and because a path for the unfiltered release of fission products could exist through the mechanical vacuum pump attached to the condenser. A rod reactivity worth of 2.5% Δk, the highest rod

worth permitted by the Technical Specifications, was assumed in the analysis. This reactivity addition would result in a peak fuel energy density of about 220 cal/gm (average across the peak fuel pellet). Perforation of the cladding of about 330 fuel rods is predicted from analyses reported in the FSAR. These analyses indicate that the calculated total excursion energy would be distributed in the core such that 330 fuel rods would have energy densities greater than 170 cal/gm, which is estimated to be the threshold energy density required to cause fuel cladding damage.

We have evaluated the consequences of the control-rod-drop accident assuming that 330 fuel rods fail, releasing 100 percent of the noble gases and 50 percent of the halogens from the affected rods to the primary system. Of the fission products released from the affected rods, all of the noble gases and 10 percent of the halogens are assumed to flow through the main steamlines to the condenser where half of the transported halogens are assumed to be removed by plateout. The released fission products entrained in the steam flow will trip the main steamline radiation detectors closing the main steam line isolation valves, shutting down the condenser mechanical vacuum pump, and closing the isolation valve downstream of this mechanical vacuum pump. The halogens reaching the condenser are assumed to be released by leakage to the atmosphere at the rate of 0.5% of the condenser volume per day. A ground level release with dilution in the wake of the building was

TABLE 9.0

PILGRIM NUCLEAR POWER STATION

OFFSITE DOSES (REM) FROM DESIGN BASIS ACCIDENTS

POSTULATED ACCIDENT	Exclusion Zone(1) 2 hours		Low Population Zone(2) Course of Accident(3)	
	Whole Body	Thyroid	Whole Body	Thyroid
Loss-of-Coolant(4)	13	110	3	70
Refueling	< 2	20	< 1	2
Control Rod Drop	< 1	3	< 1	< 1
Main Steam Line Break	< 1	8	negligible	< 1

(1) The Exclusion Zone distances for the accidents were:

Loss-of-coolant and refueling accidents = 330 meters, NW of stack
Control rod drop and steam line break accidents = 450 meters, SSW of reactor building

(2) The Low Population Zone distance - 6,840 meters (4.25 miles)

(3) The course of accident is taken as 30 days, 2 hrs, 24 hrs, and 2 hrs respectively in the Low Population Zone analyses.

(4) The primary containment is assumed to leak at an accident leak rate of 1.25 weight percent of contained atmosphere per 24 hours for the duration of the accident.

with Pasquill Type F conditions and a wind speed of two meters per second for the two-hour doses at the site boundary. At the Low Population Zone distance, these conditions were assumed to continue for eight hours, followed by spreading of the plume into a 22-1/2 degree sector for 8 to 24 hours. For this accident, the 24-hour time interval is the full course of the accident.

For the Exclusion Zone analysis, the highest calculated two hour doses were found to occur at the site boundary 450 meters SSW of the reactor building. The calculated doses resulting from exposure for two hours at this site boundary would be 3 rem to the thyroid and less then 1 rem whole body. At the Low Population Zone distance calculated doses for the course of the accident would be less then 1 rem thyroid and less than 1 rem whole body.

9.5 Steamline Break Outside Containment

The break of a main steam line outside of the drywell represents a potential escape route for reactor coolant from the vessel to the atmosphere without passage through the standby gas treatment system.

This escape route would exist only for the few seconds required for the isolation control instrumentation to sense the break and close the main steamline isolation valves.

The occurrence of a main steamline break outside containment would be sensed by either high steam flow or increased temperature in the ventilation exhaust ducts for the main steam turbine and turbine casement. The steamline isolation valves would start to close within 0.5 seconds after the steamline break is sensed. The applicant has provided analyses to show that fuel rod cladding perforations would not occur as a result of a steamline break if the isolation valve closure times, including instrument delay, are less than 10.5 seconds. To provide additional margin to assure that cladding perforations will not occur during the transient before the valves are closed and to reduce the amount of radioactivity released, the Technical Specifications require a valve closure time of not greater than 5 seconds.

The meteorological considerations assumed for this accident are the same as for the control-rod-drop accident. In our analysis, the mass of primary coolant released (30,000 lbs in 5 seconds) is assumed to have a total iodine fission product specific activity of 20 microcuries per cubic centimeter, which is the maximum coolant activity permitted by the Technical Specifications.

As in the case of the control rod drop accident, the highest calculated 2 hour doses for the Exclusion Zone analysis occurred at the site boundary 450 meters SSW of the reactor building. The calculated doses resulting from exposure for two hours at the Exclusion Zone distance were 8 rem to the thyroid and less than 1 rem whole body. At the Low

Population Zone distance, exposure for the course of the accident would result in calculated doses of less than 1 rem to the thyroid and a negligible whole body dose.

9.6 Conclusion

On the basis of our evaluation, the calculated potential radiological doses that might result from any of the design basis accidents are well within the guideline values given in 10 CFR Part 100, and are acceptable.

10.0 DESIGN BASES FOR STRUCTURES AND EQUIPMENT

The applicant divided the plant structures and equipment into two categories related to safety:

The Class I features include structures (e.g., primary containment vessel, reactor building and 400-foot-high stack) and equipment (e.g., reactor pressure vessel and internals, primary coolant system and the emergency core cooling system) whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the facility and the removal of decay heat. Class II structures and equipment are defined as those which are necessary for station operation but are not essential to a safe shutdown. We have reviewed the applicant's classification of structures and equipment and we conclude that they have been appropriately classified.

In evaluating the structural design of the Class I structures, systems and equipment, our consultant (M.S. Agbabian and G.S. Young), whose report is attached as Appendix E, considered the following: (1) the geology and nature of the foundation; (2) criteria for design loads, load combinations and design stresses; and (3) the seismic design criteria and methods of analysis.

The general requirements for the structural design of all Class I structures and equipment include provisions for resisting dead, live and operating loads together with the operating basis earthquake (OBE) (i.e., peak ground acceleration of 0.08g) within the allowable stress requirements of local and state building codes, the Uniform Building Code, the ASME[1/] Boiler and Pressure Vessel Code, the U.S.A. Standards B31.1.0 Piping Code, the AISC[2/] and the ACI[3/] Code. The Class I structures and equipment are also designed to withstand dead, live and operating loads together with the design basis earthquake (DBE) (i.e., peak ground acceleration of 0.15g) so that the resulting stresses or strains would not be excessive to cause a loss of function of equipment necessary to assure a safe shutdown of the reactor or cause a significant release of radioactivity.

[1/] American Society of Mechanical Engineers

[2/] American Institute of Steel Construction

[3/] American Concrete Institute.

In order to predict the most conservative value of force inputs in the dynamic analyses of Class I structures and equipment over a range of natural frequencies, earthquake response spectra were based on the time-history of the Taft Earthquake (1952). We and our consultants conclude that the seismic design and criteria used for the analysis of the final plant design are satisfactory.

Class I structures subject to winds have been designed to withstand loads from wind, rain, snow and ice storms within normal code allowable stresses (AISC for structural steel and ACI for reinforced concrete). We have also reviewed the effects of wind loadings and the potential missiles resulting from tornadoes. With the exception of the 400-foot-high chimney, Class I structures housing Class I equipment including the turbine building, are designed to withstand without failure a tornado wind loading of the order of 300 mph tangential wind speed, and a 3 psi pressure differential. Failure of the chimney will not preclude a safe shutdown of the station. We conclude that structures and equipment necessary for a safe plant shutdown have been adequately designed against the effects of tornadoes.

11.0 QUALITY ASSURANCE

Our review of the Pilgrim Station Quality Assurance (QA) program was based on the information presented in Appendix D of the FSAR, the response to questions regarding its implementation and results, and

information obtained from the inspection program of the Commission's Division of Compliance. The following areas were emphasized during our evaluation:

a. The qualifications and organization of the groups responsible for QA within the participating organizations.

b. The activities conducted by the applicant in implementing the QA program.

c. The programs for vendor audits, document control and retention, quality control at the site, review and approval of the various preoperational, startup, and post-operation procedures.

d. The effectiveness of the overall QA program as indicated by reports of inspections by the Division of Compliance.

e. The provisions for continuance of the QA program during operation of the station.

Based on the aforementioned evaluation and subject to a favorable final report by the Division of Compliance, we conclude that the Pilgrim Station QA program has provided reasonable assurance that the quality of this facility is acceptable.

12.0 EMERGENCY PLANNING

The applicant has prepared an Emergency Plan that would be followed in the event of either an onsite or offsite emergency situation. The plan

describes the emergency organization, assignments of authority and responsibilities of plant personnel and offsite support groups in the unlikely event of an accident that might affect the general public. Detailed procedures to protect the onsite personnel health and the safety of the general public are an integral part of the Pilgrim Operations Manual. Arrangements to deal with radiological emergencies are being made with the responsible state, local and Federal agencies. The agencies involved include the Massachusetts State Police, Massachusetts Department of Public Health and Nuclear Incident Advisory Team, Plymouth Fire Department and Plymouth Police Department. In addition, technical assistance is available through the Radiological Emergency Assistance Team Program of the AEC. The arrangements made relate to coping with various levels of emergencies to assure proper notification of authorities and initiation of the proper actions.

Members of the applicant's onsite staff will cooperate with state and local officials in providing technical advice concerning the potential offsite effects throughout the course of any accidents affecting the general public. The applicant possesses the capability for providing offsite monitoring. Adequate emergency communications and associated auxiliary power sources have been installed to provide uninterrupted liaison between onsite personnel and offsite support groups and agencies in the event of an accident.

The applicant has made arrangements with the Jordan Hospital, Plymouth, Massachusetts for receiving and treating contaminated or exposed persons requiring immediate hospital care. In addition, the applicant has made arrangements with the Shriners Hospitals for Crippled Children, Burns Institute - Boston Unit for providing extended specialized treatment of persons having radiation injuries.

We have concluded that the arrangements made by the applicant to cope with the possible consequences of accidents at the site are both reasonable and prudent, and that there is adequate assurance that such arrangements will be implemented satisfactorily in the unlikely event that they are needed. The detailed emergency plan is being reviewed by the Division of Compliance.

13.0 CONDUCT OF OPERATIONS AND TECHNICAL QUALIFICATIONS

13.1 Station Organization and Staff Qualifications

Approximately 75 full-time employees will be assigned to the station during commercial operation under the supervision of the Pilgrim Division Head who reports through the Steam Operations Department Head to the Vice President, Director of Operations and Engineering, Boston Edison. The operations staff, technical support, maintenance, and administrative support groups report to the Pilgrim Division Head through the Assistant Division Head.

The Chief Operating Engineer, who will be licensed as a Senior Reactor Operator (SRO) prior to initial fuel loading, has the responsibility for about 33 members of the operating staff, consisting of 5 shift complements of 6 men. The shift complement for unit operation will include a Nuclear Watch Engineer (SRO), a Nuclear Operating Supervisor licensed as a reactor operator (RO), three nuclear plant operators, one of which is a RO, and a radiation protection technician. A five man minimum crew size for reactor operation is required by the Technical Specifications. Supervision of a Technical Support Group (about 11 men) is the responsibility of the Chief Technical Engineer. This group is responsible for nuclear, thermal and mechancial engineering, radiation protection and surveillance, chemical engineering, and compliance. The Chief Maintenance Engineer is in charge of a Maintenance Group (about 26 men) that is responsible for all maintenance including control instrumentation, electrical power, and general station equipment.

During the station startup program, the normal operating staff will be supplemented with about 25 engineers and technicians to provide support in training, chemistry, reactor programming, instrumentation and controls, and maintenance. General Electric will also assign a startup team consisting of about 11 experienced reactor operators, engineers and technicians to the Pilgrim Station during reactor startup and power ascension tests.

We have concluded that the organizational structure for the Pilgrim Station is adequate to provide continued engineering support and an operations staff capable of operating the proposed facility safely during normal and abnormal conditions, including station startup.

The qualifications of all operating staff personnel throughout the service life of the plant will meet or exceed those defined in the ANS proposed standard for "Selection and Training of Personnel for Nuclear Power Plants" (July 3, 1969). The Assistant Division Head and members of the operating staff will be licensed operators. The Assistant Division Head, the Chief Operating Engineer and the Nuclear Watch Engineers will hold senior Reactor Operator licenses, while the Operating Supervisors and some Plant Operators will hold Reactor Operator Licenses. We find the qualifications of the Pilgrim staff acceptable.

13.2 Operating Procedures

Detailed written procedures will be prepared for the Pilgrim Station in the form of an Operations Manual. These procedures will apply to all systems and components involving nuclear safety and will comply with the Technical Specifications. Areas to be covered include: administration, power operations, maintenance and testing, fuel handling, radiological emergencies, and radiation protection.

All original procedures and changes thereto will be reviewed by the Operation Review Board and approved by the Pilgrim Division Head. The Division of Compliance also reviews certain procedures.

We conclude that the provisions for preparation, review and approval of written procedures are satisfactory.

13.3 Test and Startup Program

The test and startup program implementation is the responsibility of Boston Edison and consists of construction tests, preoperational and acceptance tests, and reactor startup and power tests. Construction tests are performed to satisfy requirements of applicable codes and to provide assurance that construction is complete in accordance with drawings and specifications. Preoperational and acceptance tests include the tests, calibrations and adjustments performed on station components, subsystems and systems to demonstrate their satisfactory operability to permit safe loading of fuel. The reactor startup and power tests are performed to demonstrate that the station is capable of operating safely and satisfactorily.

The Pilgrim Division will be assisted during these programs by the Boston Edison Nuclear Projects staff. Both staffs review the test procedures which are also approved by both the Pilgrim Division Head and the Nuclear Projects Manager of Boston Edison.

Test results of each major plateau of the startup and power test program will be reviewed and approved by the Pilgrim Division Head, Boston Edison Nuclear Projects Manager, General Electric Site Manager and the Bechtel Lead Startup Engineer, where appropriate, prior to proceeding to the next major plateau of the test program.

We have concluded that the test and startup program described by the applicant will provide an adequate basis to confirm the safe operation of the facility and is acceptable.

13.4 Safety Review and Audit

The safety review and audit functions will be conducted by the Operations Review Committee (ORC) and the Nuclear Safety Review and Audit Committee. The ORC is advisory to the Pilgrim Division Head. Membership includes the Pilgrim Division Head, Assistant Division Head, Chief Operating Engineer, Chief Technical Engineer, and Chief Maintenance Engineer. The Nuclear Safety Review and Audit Committee provides corporate management with a review and audit capability to verify that organizational checks and balances are functioning to assure continued safe operation and design adequacy of the station. The responsibilities and authorities for these committees are delineated in the Technical Specifications. We conclude that the review and audit structure proposed by the applicant is acceptable.

14.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the AEC. We have reviewed the applicant's proposed Technical Specifications in detail and have held numerous meetings with the applicant to discuss their contents. Modifications to the proposed Technical Specifications submitted by the applicant were made to describe more clearly the allowed conditions for plant operation. The finally approved Technical Specifications are part of the proposed operating license. Included are sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features and administrative controls. On the basis of our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of 10 CFR Part 20 limits. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features for continued plant operation will be available in the event of malfunctions within the plant.

15.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

The ACRS reported on the results of its construction permit review of the Pilgrim Nuclear Power Station in a letter dated April 12, 1968. We consider that the applicant has been responsive to the recommendations

of the ACRS as indicated in this letter, and we conclude that the matters raised have been satisfactorily resolved during the design and construction of the Pilgrim Station.

The ACRS completed its review of the application for an operating license for the Pilgrim Station during its 132nd meeting held April 1-3, 1971. A copy of the ACRS letter, dated April 7, 1971, is attached as Appendix A.

The ACRS, in its letter, made several recommendations and noted several items to be resolved by the applicant and the staff either before plant operation or on an acceptable time scale subsequent to initial operation. These items have been considered in our evaluation and include:

The design of the liquid waste concentrator system should reduce to levels as low as practicable the amount of long-lived radioisotopes discharged to the bay, (Section 8.2); the operating mode of the liquid radwaste system before and after the installation of the waste concentrator should be consistent with reducing the radioactivity released (Section 8.2); the gaseous radwaste system should be modified to provide a much lower limit on gaseous and particulate radioactivity discharged through the stack (Section 8.3); the primary containment should be inerted during normal operation, and the primary control of accident-generated hydrogen should be by some method other than purging (Section 4.1.2); the applicant should protect the containment against

breaching that may be caused by whipping of unrestrained piping in the event of a pipe rupture (Section 3.3.4); the applicant should guard against missiles that could be generated from the biological shield by rupture of pipes, including safe-ends, within the shield (Section 3.3.4); the applicant should assure by design modification if necessary, that the accidental dropping of the spent-fuel cask into the fuel pool will not cause leakage in excess of the make-up capacity (Section 7.2); the applicant should make tests to confirm the predicted vibrational characteristics of the reactor vessel internals (Section 3.3.3); the applicant should make timely proposals for resolution of the problem of possible failure to scram on anticipated transients (Section 6.2); the applicant should reevaluate, before routine operation at full power, the performance of the emergency core cooling system, using recent heat transfer data and calculational methods (Section 5.8); several items regarding plant instrument systems and electrical power systems that were under review by the Regulatory Staff should be resolved (Sections 6.3, 6.4, and 6.5); the applicant should continue to explore means of improving access to vessel wells for inservice inspection (Section 3.3.8); the reactor vessel pressure should be limited in accordance with current AEC bases when the vessel temperature is below 180°F (Section 3.3.2); and the applicant should explore the feasibility of using an alternative 23 kV supply or of making local changes to reduce the possibility of losing the 345 kV and 23 kV offsite power lines simultaneously (Section 6.6).

The applicant has agreed to implement the recommendations of the ACRS. We will follow implementation of the recommendations of the ACRS during operation of the facility. The ACRS concluded in its letter that if due regard is given to the items mentioned above, the Pilgrim Station can be operated at thermal power levels up to 1998 MWt without undue risk to the health and safety of the public.

16.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all the directors and principal officers of the applicant are citizens of the United States.

The applicant is not owned, dominated or controlled by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data that might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons, and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

17.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to the financial data and information required to establish financial qualifications for an applicant for operating licenses are 10 CFR 50.33(f) and 10 CFR 50 Appendix C. Boston Edison Company's application as amended by the letter of James M. Carroll, Vice President and General Counsel, dated February 19, 1971, and the accompanying certified annual financial statements provided the financial information required by the Commission's regulations.

These submittals contain the estimated average annual operating cost of the Pilgrim facility plus the estimated cost of permanently shutting down the facility and maintaining it in a safe shutdown condition. The estimated operating costs per year on the average will be $38 million. Such costs include carrying charges on plant investment, operating and maintenance, insurance and fuel. The applicant's estimate of the cost of permanently shutting down the facility is $1 million and $200,000 annually thereafter for maintaining it in a safe condition.

We have examined the certified financial statements of Boston Edison Company to determine whether the Company is financially qualified to meet these estimated costs. The information contained in the 1970 financial report indicates that operating revenues for 1970 totaled $219.8 million; operating expenses were $182.3 million; the interest on the long-term debt was earned 2-1/2 times; and the net income for the

year was $26.9 million, of which $19.3 million was distributed as dividends to the stockholders and the remainder of $7.6 million was retained for use in the business. As of December 31, 1970, the Company's assets totaled $789.1 million, most of which was invested in utility plant ($714.6 million), and earnings reinvested in the business were $84.6 million. Financial ratios computed from the 1970 statements indicate a sound financial condition, e.g., long-term debt to total capitalization -.52, and to net utility plant - .46; net plant to capitalization - 1.15; the operating ratio - .83; and the rates of return on common - 11.4%, on stockholders' investment - 9.1%, and on total investment - 6.1%. The record of the Company's operations over the past 5 years reflects that operating revenues increased from $166.5 million in 1966 to $219.8 million in 1970; net income increased from $21.1 million to $26.9 million; and net investment in plant from $575.1 million to $717.1 million. Moody's Investors Service rates the Company's first mortgage bonds as Aa. The Company's current Dun and Bradstreet rating is AaA1.

Our evaluation of the financial data submitted by the applicant, summarized above, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR 50.33(f) with respect to the operation of the Pilgrim Nuclear Power Station. A copy of the staff's financial analysis is attached as Appendix F.

18.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licenses for facilities such as power reactors licensed under 10 CFR Part 50.

18.1 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The Boston Edison Company is, with respect to Pilgrim Nuclear Power Station, subject to the foregoing requirements, and has taken the following steps with respect thereto.

The Company has timely furnished to the Commission proof of financial protection in the amount of $1,000,000, in the form of a Nuclear Energy Liability Insurance Association policy (Nuclear Energy Liability Policy, Facility Form) No. NF-188.

Further, the Company executed Indemnity Agreement No. B-48 with the Commission as of November 20, 1970, the effective date of its pertinent preoperational fuel storage license, SNM-1193. The Company has paid the annual indemnity fee applicable to preoperational fuel storage.

18.2 Operating License

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from, for example, preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for reactors which have a rated capacity of 100,000 electrical kilowatts or more is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently $82 million.

Accordingly, no license authorizing operation of PNPS will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended that the policy limits have been increased to the $82 million figure to meet the requirements of the Commission's regulations for reactor operation. Similarly, no operating license will be issued until an appropriate amendment to the present indemnity agreement has been executed. Boston Edison Company will be required to pay an annual fee for operating license indemnity as provided in our regulations, at the rate of $30 per each thousand kilowatts of thermal capacity authorized in any operating license.

18.3 Conclusion

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the provisional operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement or amendment thereto with the Commission.

19.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have concluded that:

1. The application for facility license filed by the Boston Edison Company, as amended, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and

2. The construction of the Pilgrim Nuclear Power Station (the facility) has proceeded, and there is reasonable assurance that it will be completed in conformity with Provisional Construction Permit No. CPPR-49, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

3. The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

4. There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1; and

5. The applicant is technically and financially qualified to engage in the activities authorized by this operating license in

accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and

6. The applicant has furnished proof of financial protection to satisfy the applicable requirements of 10 CFR Part 140; and

7. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.

Prior to any public hearing on the matter of the issuance of an operating license to the Boston Edison Company for the Pilgrim Nuclear Station, the Commission's Division of Compliance will prepare a report which will deal with those matters relating to the status of construction completion and conformance of that construction to the construction permit and the application. Before an operating license will be issued to Boston Edison Company, assuming such a license is authorized following the public hearing, the facility must be completed in conformity with the construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Division of Compliance prior to issuance of a license.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 7, 1971

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: PILGRIM NUCLEAR POWER STATION

Dear Dr. Seaborg:

At its one-hundred thirty-second meeting, April 1-3, 1971, the Advisory Committee on Reactor Safeguards reviewed the application by Boston Edison Company for authorization to operate the Pilgrim Nuclear Power Station, comprising a single nuclear power generating unit, at power levels up to 1998 MW(t). The application was previously considered at a Subcommittee meeting held at the site on March 22, 1971. During its review the Committee had the benefit of discussions with representatives and consultants of Boston Edison Company, General Electric Company, Bechtel Corporation, and the AEC Regulatory Staff, and of the documents listed below. The Committee reported to you at the construction permit stage for this station on April 12, 1968.

The Pilgrim Nuclear Station is on the west shore of Cape Cod Bay, approximately five miles from the center of Plymouth, Massachusetts (population about 11,000). Boston is 36 miles northwest of the site, and Providence is 44 miles west. The Pilgrim reactor is a boiling water reactor generally similar to Millstone Unit 1 and other boiling water reactors recently reviewed by the Committee for operation.

The applicant has not provided equipment for concentrating and separating radioactivity from liquid wastes, and he states that the radioactivity concentration in the condenser circulating water discharge will not exceed that permitted by 10 CFR 20. During the first reactor shutdown for refueling, the applicant will install an evaporator designed to permit the holdup of liquid wastes and thereby reduce the gross radioactivity discharged. The Committee believes that the design and operation of this evaporator system should be such as to reduce to levels as low as practicable the amount of long-lived radioisotopes discharged. The Regulatory Staff should review and approve the design

and operating mode of this equipment. The Committee also believes that prior to the installation of this equipment, effort should be made to reduce the radioactivity released.

The applicant proposes that the gaseous and particulate radioactivity discharged through the stack will not exceed 10 CFR 20 limits. The Committee believes the applicant should set a much lower operating limit and should make such equipment changes as may be necessary to accomplish this.

In previous reports, the Committee has commented on the following matters common to boiling water reactors recently reviewed for operation; these comments apply also to the Pilgrim Plant. The Committee believes that the reactor containment should be inerted during normal operation, and that the primary control of accident-generated hydrogen should be by some method other than purging; the need for inerting should be reviewed periodically as operating experience and further knowledge from development work are obtained and as other means of coping with hydrogen are found. The applicant proposes to protect the containment against breaching that may be caused by whipping of unrestrained piping in the event of a pipe rupture, and also to guard against missiles that could be generated from the biological shield by rupture of pipes, including safe-ends, within the shield.

The applicant proposes to assure that accidental dropping of the spent-fuel cask into the fuel storage pool will not cause leakage in excess of the make-up capacity, and will make such modifications as may be necessary. The applicant said he would make tests adequate to confirm the predicted vibrational characteristics of the vessel internals. The Committee believes the applicant should make timely proposals for resolution of the problem of possible failure to scram on anticipated transients. The applicant should reevaluate, before routine operation at full power, the performance of the emergency core cooling system, using recent heat transfer data and calculational methods. Several items regarding plant instrument systems and electrical systems are under review by the Regulatory Staff. All these matters should be resolved to the satisfaction of the Regulatory Staff; the Committee wishes to be kept informed.

The Committee believes the applicant should continue to explore means of improving access to vessel welds for inservice inspection. The Committee also believes that the reactor vessel pressure should be limited in accordance with current AEC bases when the vessel temperature is below 180°F.

The site is served by two 345 kv electrical transmission lines on the same towers and a separate 23 kv line. Over a short distance, the lines are adjacent and it is physically possible for the fall of a tower to break the 23 kv line. The Committee believes that the applicant should explore the feasibility of using an alternative 23 kv supply or of making local changes to reduce the possibility of losing the 345 and 23 kv lines simultaneously.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Pilgrim Nuclear Power Station can be operated at power levels up to 1998 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H. Bush

Spencer H. Bush
Chairman

References - Pilgrim Nuclear Power Station

1. Amendment Nos. 12 through 27 to License Application for the Pilgrim Nuclear Power Station and Volumes 1 through 5 of FSAR.

2. Letter from Boston Edison Company, dated September 26, 1968; Report entitled "Installation of Stub Tubes - Pilgrim Station Reactor Vessel - Boston Edison Project".

APPENDIX B

Comments on

Pilgrim Nuclear Power Station
Final Safety Analysis Report
Volumes I through V dated December 31, 1969

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
June 2, 1970

Onsite meteorological statistics indicate that at a level of 220 feet above the surface (300 feet Mean Sea Level) the annual mean wind speed is 16 mph and the frequency of calms is 0.43 percent. At this level, stable (moderate and slight) conditions with onshore winds and a speed of 3 mph or less (including all calms) occurs less than 1 percent of the time. Consequently, it would be reasonable to conclude that a diffusion factor (χ/Q) equal to or greater than that equivalent to Pasquill Type F and a wind speed of 2 m/sec would occur less than 5 percent of the time.

For the elevated releases from a stack which is 100 m above the surface, the mechanism resulting in the highest ground concentration is either a rapid mixing downward or a decrease in the effective stack height because of rising downwind terrain. The nearest site boundary, other than the shoreline, is at a distance of 330 m west northwest of the stack where the terrain is about 90 m below the top of the stack. In this situation a very rapid diffusion (Pasquill Type A) with a wind speed of 1 m/sec and an effective stack height of 90 m would result in a relative concentration of 2×10^{-5} sec m^{-3}. The highest nearby terrain is toward the south southwest at a distance of 800 m and at 2200 m. In the former (Roadside Park on site boundary) the terrain is 40 m below the stack and in the latter (Manoment Hill) it is at the same elevation as the stack top. In this situation the controlling condition seems to be an inversion condition with zero effective stack height at the distance of Manoment Hill and a wind speed of 2 m/sec resulting in a concentration of 1×10^{-4} sec m^{-3}.

The site investigation of sea breezes (Appendix I) verifies another mechanism, that of fumigation, which brings an elevated plume rapidly to the surface. In the case of the nearest site boundary we have assumed the initial plume to remain horizontal despite underlying terrain, the wind speed to be 2 m/sec, the horizontal spread to be equivalent to Type F and the mixed layer to be 90 m thick. The resulting concentration is 2×10^{-4} sec m^{-3}.

For a surface release, the controlling condition would be an inversion (Type F) with a 2 m/sec wind speed. Allowing a factor of 3 for the aerodynamic effect of the building complex, the resulting concentration at the nearest site boundary of 330 m is 1.4×10^{-3} sec m^{-3}.

2

The controlling onshore direction with regard to the average annual concentration appears to be a wind from the north northeast towards the high terrain (Roadside Park) at the site boundary. Assuming an effective stack height of 40 m, a 6 m/sec wind speed and a frequency of 1.5 percent each of unstable (Type B) and neutral (Type D) and a 1.0 percent frequency of inversion (Type F) conditions (see Tables 2.3.3 to 2.3.6), the resulting average concentration over a $22\frac{1}{2}^{\circ}$ sector at 800 m was 1.4×10^{-7} sec m^{-3}.

In summary, for the cases listed we compute the controlling concentrations at offsite locations to be as follows:

Elevated, 0 - 2 hr release	:	2×10^{-4} sec m^{-3}
Ground, 0 - 2 hr release	:	1.4×10^{-3} "
Elevated, annual release	:	1.4×10^{-7} ".

The latter is in good agreement with the applicant's Table E.0.16.

IN REPLY REFER TO:

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON, D. C. 20240

APR 27 1970

Mr. Harold L. Price
Director of Regulation
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Mr. Price:

These are our comments on the application by the Boston Edison Company for an operating license for the Pilgrim Nuclear Power Station, Plymouth County, Massachusetts, AEC Docket No. 50-293. In response to the letters of January 16 and March 3, from your Mr. Boyd, Assistant Director for Reactor Projects, Division of Reactor Licensing, the Fish and Wildlife Service has reviewed the Final Safety Analysis Report, including Amendment No. 13, which provides information on project operations.

The applicant's proposed environmental radiological and ecological monitoring program is in general agreement with recommendations included in our letter of December 11, 1967. This monitoring program was developed in cooperation with the Fish and Wildlife Service and the Massachusetts Department of Natural Resources and will be conducted under the supervision of an Administrative-Technical Committee, which includes four representatives of the Service.

The applicant has given assurance that pre-operational and post-operational environmental surveys will be conducted according to the recommendations of the advisory committee and that public recreational use facilities will be installed at the project site.

We are concerned with the possibility of damages to aquatic life by the heated effluent, chemicals used to control growth in the condensers, and the destruction of aquatic organisms in the cooling water as it passes through the condenser system. Though our review of project plans indicates that construction and operation of this project should not present any unresolvable environmental problems, we urge that before the operating license is issued, the Boston Edison Company be required to give the additional assurance that modifications in project structures and operations for the protection of fish and wildlife resources deemed necessary by the Administrative-Technical Committee will be incorporated into the project design.

Sincerely yours,

Commissioner

APPENDIX D
- 125 -

BOSTON EDISON COMPANY

EXECUTIVE OFFICES
800 BOYLSTON STREET
BOSTON, MASSACHUSETTS 02199

JAMES M. CARROLL
VICE PRESIDENT AND
GENERAL COUNSEL

RECEIVED
FEB 4 1971
U.S. ATOMIC ENERGY COMMISSION
Regulatory Mail Section

February 1, 1971

Mr. Peter A. Morris, Director
Division of Reactor Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Re: Docket No. 50-293

Dear Mr. Morris:

By your letter of January 21, 1971, you called our attention to the recommendations of the Fish and Wildlife Service as they relate to the Pilgrim Nuclear Power Station and requested that we cooperate with the Administrative-Technical Committee of the State of Massachusetts in providing modifications in project structures and operations for the protection of fish and wildlife resources should they be deemed necessary.

Should the Commonwealth of Massachusetts deem it necessary, we will provide modifications in project structures and operations for the protection of fish and wildlife resources. Although the function of the Administrative-Technical Committee is advisory in nature, we are confident that their review and comments will be reflected in any recommendation made by the Commonwealth.

Sincerely yours,

James M. Carroll

James M. Carroll

cc: Mr. Claude A. Pursel

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AJA

R-6821-1843

Final Report

A REVIEW OF THE ADEQUACY OF THE STRUCTURAL DESIGN CRITERIA FOR THE PILGRIM NUCLEAR POWER STATION

March 19, 1971

(Docket No. 50-293)

Prepared for

U.S. ATOMIC ENERGY COMMISSION
Division of Reactor Licensing
Washington, D.C. 20545

by

M. S. Agbabian
G. A. Young

INTRODUCTION

PURPOSE AND SCOPE

This report provides a review of the adequacy of the structural design criteria for the Pilgrim Nuclear Power Station. It is based upon the Final Safety Analysis Reports (FSAR) with Amendments 1 through 20 which have been submitted in support of an application to the Atomic Energy Commission (AEC) for a facility operating license. The applicant is the Boston Edison Company. Reference has also been made to the Preliminary Safety Analysis Reports and amendments and the Safety Evaluation Report of 1968 for the same facility. All documents used are listed under References at the end of this report.

The scope of this review is restricted to a consideration of the seismic design criteria which establish the survivability and operability of the Class I structures and equipment when subjected to the design earthquake and the criteria which ensure safe plant shutdown and containment following a maximum credible earthquake.

SITE AND ENVIRONS

The site contains slightly over 500 acres and is located in the town of Plymouth, Massachusetts on the western shore of Cape Cod Bay.

GEOLOGY AND SEISMOLOGY

Very thorough and complete geological and seismological investigations have been conducted of the site and surrounding region. The site is about 14 to 32 ft above mean sea level. Bedrock is about 64 ft below mean sea level and is covered by glacial and recent deposits. An upper discontinuous, erratic zone of sandy silts, and silty and clayey sands was disclosed up to 20 ft in thickness. Material of questionable density from this layer was removed and replaced with well compacted granular material. Materials below a depth of 35 ft are compact and dense.

There are no known faults at or near the site and none were revealed by the drilling or geophysical investigations. The most seismically significant structure features are located about 60 miles north of the site. The seismic investigation indicates that the site is not an active seismic area. A design earthquake intensity of V and a maximum credible earthquake intensity of VII were assigned as a result of this investigation. We are in agreement with the scope and detail of the geological and seismological investigations and with the earthquake intensities assigned.

SEISMIC CRITERIA

Based upon the geological and seismological investigations, a peak ground acceleration of 0.08 g was assigned for the design of all Class I structures and a peak acceleration of 0.15 g was designated and used for safe shutdown design. Response spectra were developed with different percents of critical damping which were consistent with these peak accelerations. We have concluded that the response spectra used for design are reasonable and conservative.

BASES OF DESIGN

The seismic design of Class I structures and equipment have been reviewed. Class I structures and equipment are those whose failure or malfunction might cause or increase the severity of an accident which would endanger the public health and safety. This category includes those structures, equipments, and components required for safe shutdown and isolation of the reactor.

CLASS I STRUCTURES

The seismic loads, combinations of loads, load factors or factors of safety, and methods of dynamic analysis have been examined carefully. We concur with the procedures used by the applicant in the design of the Class I structures for seismic loads.

CLASS I EQUIPMENT

In order to analyze Class I equipment, the applicant used the Taft earthquake record as input to a mathematical model of the structure. Response spectra were then generated for each floor level for the seismic analysis of the equipment. For frequencies below 1 cps, no response lower than given by the basic criteria ground response spectra was used. We concur with this procedure. The seismic loads, combination of loads, factors of safety, and methods of analysis used have been reviewed. We agree with the procedures used by the applicant in the design of the Class I equipment.

CONCLUSION

After review of the FSAR, amendments thereto, and other material made available to us, we make the following statement as to the adequacy of the seismic design of the Pilgrim Nuclear Power Station. The design is considered to be adequate in terms of provisions for safe shutdown for a design basis earthquake of 0.15 g maximum horizontal ground acceleration and to withstand otherwise the effects of an earthquake of one-half this magnitude. On the basis of the information made available to us, we conclude that the design incorporates margins of safety which are acceptable.

REFERENCES

1. *Final Safety Analysis Reports*, Vols. I through V, Pilgrim Nuclear Power Station, 1970 and Amendment Nos. 14, 16, 17, 18, 19, 20, and 21.

2. *Design and Analysis Report*, Vols. I, II, III, Pilgrim Nuclear Power Station, 1967 and Amendment Nos. 1 through 10.

3. *Safety Evaluation by The Division of Reactor Licensing, U.S. Atomic Energy Commission*, Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, Massachusetts, Docket No. 50-293, May 20, 1968.

Mihran S. Agbabian

Mihran S. Agbabian

George A. Young

George A. Young

APPENDIX F

BOSTON EDISON COMPANY
DOCKET NO. 50-293
FINANCIAL ANALYSIS

	(dollars in millions) Calendar Year Ended Dec. 31		
	1970	1969	1968
Long-term debt	$ 325.8	$ 266.2	$ 256.8
Utility plant (net)	714.6	594.3	526.4
Ratio - debt to fixed plant	.46	.45	.49
Utility plant (net)	714.6	594.3	523.0
Capitalization	622.7	515.5	498.5
Ratio - net plant to capitalization	1.15	1.15	1.05
Stockholders' equity	296.9	249.3	241.6
Total assets	789.1	652.1	577.2
Proprietary ratio	.38	.38	.42
Earnings available to common	24.4	23.1	21.7
Common equity	214.0	206.3	198.7
Rate of return on common equity	11.4%	11.2%	10.9%
Net income	26.9	25.1	23.6
Stockholders' equity	296.9	249.3	241.6
Rate of return on stockholders' equity	9.1%	10.1%	9.8%
Net income before interest	48.4	39.7	34.4
Liabilities and capital	789.1	652.1	577.2
Rate of return on total investment	6.1%	6.1%	6.0%
Net income before interest	48.4	39.7	34.4
Interest on long-term debt	19.0	12.2	8.7
Times fixed charges earned	2.5	3.3	4.0
Net income	26.9	25.1	23.6
Total operating revenues	219.8	200.2	189.3
Net income ratio	.122	.125	.125
Operating expenses (incl. taxes)	182.3	166.3	157.5
Operating revenues	219.8	200.2	189.3
Operating ratio	.83	.83	.83
Retained earnings	84.6	76.9	69.3
Earnings per share of Common	$3.31	$3.10	$2.90

	1970		1969	
Capitalization	Amount	% of Total	Amount	% of Total
Long-term debt	$325.8	52.3%	$266.2	51.6%
Preferred stock	82.9	13.3	43.0	8.4
Common stock	214.0	34.4	206.3	40.0
Total	$622.7	100.0%	$515.5	100.0%

Moody's Bond Rating:
First Mortgage Aa

Dun and Bradstreet Credit Rating AaA1

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